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Proceedings of the Twenty-Seventh Water Reactor Safety Information Meeting

Held at
Bethesda Marriott Hotel
Bethesda, Maryland
October 25–27, 1999

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research

Proceedings prepared by
Brookhaven National Laboratory

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ABSTRACT

This report contains papers presented at the Twenty-Seventh Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1999. The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from Belgium, Finland, France, Japan, Russia, and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

PROCEEDINGS OF THE
27TH WATER REACTOR SAFETY INFORMATION MEETING
OCTOBER 25-27, 1999

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Preliminary Evaluation Results from the 27th WRSM-1999

Background

The Nuclear Regulatory Commission's Office of Nuclear Regulatory Research (RES) engaged contractor assistance (1) to evaluate the effectiveness of the Water Reactor Safety Meeting (WRSM) in creating a dialogue on research and in meeting the needs and interests of stakeholders, and (2) to gain feedback on improving stakeholder involvement in the research process. We define stakeholders as groups or individuals who are affected by or who affect NRC research programs.

RES' contractor, Advanced Technologies and Laboratories, International, Inc. (ATL), conducted an on-site evaluation at the WRSM by asking attendees to provide feedback through Meeting Evaluation Forms, Technical Session Feedback Forms and one-on-one Interviews. This summary provides preliminary findings of the evaluation conducted at the WRSM. At this time, additional data is being collected through mailed surveys to external and internal stakeholders.

WRSM Evaluation Summary

The attendance at WRSM '99 totaled approximately 300 people, including 187 external stakeholders and 118 NRC pre-registrants. The conference was attended by approximately ninety non-U.S. participants, representing nineteen different countries. Of the 210 U.S. attendees, approximately forty-three representatives were from the commercial nuclear industry, thirty-five attendees were from other federal agencies, and seventeen were contractors.

Evaluation response rates at WRSM are considered a good sample for meetings. Out of approximately three hundred attendees, forty-one Interviews (14%) were conducted and forty-one Meeting Evaluation Forms were completed (14%). Responses for the Technical Session Forms ranged from nine to thirty and averaged 19.5 per session.

The stated objective of WRSM '99 was "to promote a dialogue with stakeholders in commercial nuclear applications on regulatory research that is developing and confirming technical bases for regulatory outcomes." A new format was introduced which encouraged open dialogue in facilitated sessions to identify where agreement exists on technical issues and what questions remain to be answered by research. The majority of respondents reacted favorably to the new format and were supportive of RES' goal of increasing dialogue. 74% of respondents ranked the WRSM '99 topics excellent or above average; 63% of respondents rated the new technical session format as excellent or above average; and 55% of respondents indicated that the facilitated discussion at the end of each technical session was essential or very important.

The time of year, location, length and frequency of WRSM were supported by a large majority of respondents: 86% believe the October date is appropriate; 79% believe WRSM should continue to take place annually; 89% support the present length of three days; and 97% agree that the location in the Washington, D.C. metro area is appropriate.

Three areas of suggested improvement were consistent themes in the feedback received from all stakeholder groups. Respondents supported the new format and the NRC's commitment to enhancing the opportunity for dialogue at the WRSM; many felt that this year's format should be refined and applied more consistently in each technical session to ensure that there is adequate time for quality discussion. Many respondents also suggested that the connection between research and regulatory activities be integrated into the WRSM to a greater extent. Finally, respondents encouraged broadening the representation from the utilities.

Concerning the second objective of the evaluation, i.e., improving stakeholder involvement in NRC's research process, a majority of respondents supported reviewing research ideas, submitting new research topics, providing input to RES decision-makers on research, and broadening dissemination of, and feedback on, research papers as areas in which RES stakeholders should be involved in the research process. Specifically, respondents indicated that holding meetings and publicizing the research agenda on a RES web site were the preferred communication methods to enhance stakeholder involvement in the research process. In response to this feedback, and as a first step in following through on the evaluation recommendations, the RES web page has been updated to describe FY2000 RES activities.

Next Steps

The final results of the evaluation will be available in March 2000. The final report will include an analysis of the data from all the evaluation methods. It will also include specific recommendations to RES on how to improve stakeholder dialogue at the WRSM and, how to enhance participation by stakeholders in the RES research process. The results of the analyses and the recommendations will be factored into the planning of WRSM-2000. For more information about the evaluation, please contact Isabelle Schoenfeld at iss@nrc.gov.

We would like to take this opportunity to thank all of you who participated in this evaluation; your feedback is appreciated and will lead to improvements in the WRSM and in stakeholder involvement in the research process.

OPENING REMARKS

by

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Good morning! My name is Ashok Thadani and I am the Director of the Office of Nuclear Regulatory Research at the Nuclear Regulatory Commission.

The Office of Nuclear Regulatory Research conducts independent experiments and analyses, develops technical bases for realistic safety decisions, and prepares the agency for the future by evaluating safety issues involving current and new designs and technologies. This meeting is very important in that it facilitates the exchange of information about research programs and provides an opportunity for interactions with stakeholders and professional networking, and in this light I welcome you all to the 27th Annual Water Reactor Safety Meeting (WRSM-27). Welcome to a forum designed to facilitate open dialogue and specific discussions on a range of nuclear safety research issues from both domestic and foreign perspectives. With the number and frequency of changes taking place both inside the NRC and external to the NRC, it is especially important that we fully utilize this opportunity to continue discussions of key safety issues.

I am particularly looking forward to this year's plenary sessions on "The Impact of Emerging Technologies on Nuclear Safety Research" and "How Best to Focus Both on Safety and Unnecessary Burden Reduction - The Research Role." We are fortunate to have the presence of world-class experts to guide us into more meaningful dialogue on these topics.

WRSM-27 brings together international and domestic expertise to hear progress reported and debate the degree of rigor and robustness in the scientific knowledge and technical basis for the findings and conclusions being presented. The knowledge and basis that are up for debate here are those that transcend organizations and even national boundaries. This meeting will focus on subjects such as risk-informed regulation, enhancements to regulatory effectiveness, the integrity of primary coolant pressure boundaries, the behavior of high-burnup and mixed-oxide fuels, the effects of burnup on criticality determinations, improved accident fission product source terms for operating reactors, and improved cognizance of new technologies.

During this year the Center for Strategic and International Studies (CSIS) examined the regulatory process for nuclear power plants and made recommendations throughout its report. CSIS study report and their Chairman, John Ahearne's, summary to the Commission emphasized the need for research. Current industry gains such as the technical basis for risk-informed regulation and the resolution of license renewal issues came from past NRC investments in research efforts in areas such as those used to develop Probabilistic Risk Analysis (PRA) methods beginning in the 1980s and those used to understand the effects of aging on plant systems, structures and components in the 1990s. Basically, unless sufficient resources are put into research now, future gains are not likely to be realized.

Yet, there has been a continued decline in resources which has led to the loss of research facilities, strong technical capability and leadership. This raises questions about the preparations for future challenges. With the continued decline in resources, our initiatives in cooperative research have become increasingly important to the overall regulatory research program.

The nuclear industry is recognized as a world industry and cooperative research is a cornerstone to leverage research resources and support basic nuclear capabilities. The NRC conducts research in partnership with DOE, EPRI, and more than 60 countries. Much of the research is concentrated in the areas of severe accidents, thermal-hydraulics, fuel behavior, PRAs, piping integrity and material research. Examples of how well this is done are reflected in the ROSA large-scale Test Facility in Japan for safety system testing, the work to develop advanced methods of pressure vessel fracture, and advanced predictions of pressure vessel fracture. Other examples include the Cabri Reactor in France and the NSRR reactor in Japan that are providing much of the needed data in the area of assessing fuel behavior under high burn-up accident conditions. So really, this is not just a USNRC meeting gathering of some of the most talented people in the world.

This year the meeting has a new format to promote more dialogue on safety research with the stakeholders in commercial nuclear applications. Safety research develops and confirms technical bases related to the public health and safety mission and our related strategic goals. The two themes of our meeting flow from these goals. The first theme focuses on research aimed at facilitating risk-informed regulation and reducing unnecessary burdens on industry and regulators. The focus of the second theme is to improve awareness of operating experience and our response to emerging issues.

The Office of Nuclear Regulatory Research has several self-assessment initiatives as part of the NRC's strategic performance goal of improving the effectiveness, efficiency, and realism of decisions and activities. For WRSM-27, we have engaged assistance to help us evaluate the effectiveness of this meeting and the direction the WRSMs should be going. You are the important part of the corporate wisdom for the evaluation of this meeting. We need your full participation in this evaluation as it occurs throughout the meeting. Please extend that extra effort needed to provide us your feedback. Later, the Office's Deputy Director, Margaret Federline, will discuss the changes in format for the meeting and our expectations for the evaluation.

Now, I am privileged to introduce Commissioner Jeffrey S. Merrifield, our keynote speaker. Before joining the NRC, Commissioner Merrifield had served since 1995 as Counsel and

Staff Director to the Senate Subcommittee on Superfund, Waste Control and Risk Assessment. From 1992 to 1995, he was an associate with the Washington, D.C., law firm of McKenna & Cuneo. From 1990 to 1992, he served as a legislative assistant to Senator Robert C. Smith, and from 1987 to 1990 as a legislative assistant to then Senator Gordon J. Humphrey.

A native of Antrim, New Hampshire, Commissioner Merrifield received his Bachelor of Arts degree, magna cum laude, in political science and history from Tufts University in 1985 and his Doctor of Jurisprudence degree from Georgetown University Law Center in 1992. We are privileged to have him with us today to share his thoughts on nuclear regulatory research.

KEYNOTE SPEECH

by

The Honorable Jeffrey S. Merrifield
Commissioner
U.S. Nuclear Regulatory Commission

Good Morning. Thank you very much for the opportunity to speak to you this morning. It is a pleasure and honor to be here.

Over the next few days, you will hear several presentations regarding risk including ones on the NRC's efforts to risk-inform Part 50 and our efforts to develop a risk-informed reactor oversight program. Do me a favor. If during the presentations you don't hear the word **communication**, I urge you to challenge the presenter. Ask the presenter what their plans are for communicating about risk to the lawyer on the Commission or in Congress, or to the concerned mechanic or parent who lives near a nuclear plant. Let me make one thing perfectly clear, we can have the most advanced risk insights, the best science, the leading experts in the field, but if we do not have an effective communication plan, we will fail. The only way the NRC and the nuclear industry will succeed in their efforts to risk-inform our regulations and use risk insights to reduce unnecessary burden is by learning to effectively communicate with the public and our other stakeholders about risk. For most of our stakeholders and even some of our staff, risk is an unknown, a black box. Uncertainty brings with it apprehension, fear. Other stakeholders in public interest groups and even in Congress view our efforts to risk-inform our regulations with skepticism. They see these risk initiatives as just another ploy by the industry and a capitulating NRC to reduce regulatory requirements.

It is imperative that we discuss risk in a manner that brings greater understanding and confidence to our community of stakeholders. We must demonstrate that risk-informed regulation in no way represents less of a commitment to safety, in fact, it represents an even greater commitment to safety because it allows the NRC and our licensees to focus our limited resources on aspects of plant operation that are truly risk-significant. The ball is in our court. If we are to reap the tremendous benefits of our improved risk expertise, we cannot take shortcuts in the area of communication.

State of the Nuclear Industry

What I thought I'd do today is discuss the rapidly changing state of the nuclear industry in the U.S., the equally rapid changes ongoing within the NRC, and some of the challenges facing the NRC's research and international programs.

These are truly eventful times for the electric industry in the United States. Deregulation is rapidly changing the business landscape of the electric industry and the transition to a competitive market has begun to take shape in many States. Utilities are restructuring, mergers are taking place, many nuclear and non-nuclear assets are changing hands, and the economics of the market are demanding that plant owners work very hard to control and reduce costs. The speed of change is unprecedented.

As for the state of the U.S. nuclear power industry, just a couple of years has made a tremendous difference. Just 2 years ago, the outlook for nuclear power in the U.S. appeared bleak. There were ominous predictions by both the electric industry as well as the financial community that the unpredictable regulatory and economic challenges brought on by deregulation would force many nuclear plants to shut down well before the expiration of their licenses. Today, many economic factors and the NRC's regulatory reform efforts have contributed to a resurgence for nuclear power. As I have stated on many occasions, I believe the outlook for nuclear power in the U.S. is the brightest its been since the Three Mile Island accident. Deregulation of the electric industry, once thought to be final nail in nuclear power's coffin, is instead serving as the catalyst for dramatic improvements in the manner in which nuclear plants are managed. Licensees have improved operator training, made significant process improvements, streamlined operations, shortened refueling outages, and reduced costs. Today, plants are operating better than ever before with forced outage rates at an all time low and capacity factors at an all time high. Utilities are recognizing that nuclear plants that are well-maintained and effectively operated can be money-makers even in a competitive deregulated electric industry.

The resurgence of nuclear power in the U.S. has catapulted two issues to the forefront of the nuclear industry: license renewal and license transfers.

<u>**License Renewal**</u>

As many of you know, more than half of the operating nuclear reactors in the U.S. will reach the end of their currently licensed lives by 2018. Both the NRC and the nuclear power industry have devoted extensive resources to understanding the technical issues associated with allowing a nuclear plant to operate for 20 years beyond the original 40-year license term. Also, establishing a disciplined, predictable, and timely license renewal process that ensures the protection of public health and safety, has been, and continues to be, a top priority of the Commission. In this regard, the Commission has issued a policy statement outlining its expectations for a focused review of license renewal applications. Using case specific Orders, the Commission has established an aggressive adjudicatory schedule for reviewing the first two license renewal applications - those for the Calvert Cliffs and Oconee nuclear power plants. Significant management oversight of the review process and the resources dedicated to that process has been institutionalized. As I have told the staff and a number of industry leaders, ultimately the decision on whether to seek license renewal rests with the licensee. This decision should not be tainted by concern over the uncertainty and unpredictability of the NRC's regulatory processes.

While our goal for completing the license renewal process for the Calvert Cliffs and Oconee applications was originally 30-36 months, it is now likely these reviews will be completed well ahead of schedule - most likely within 25 months. While I am personally pleased with our progress on these two applications, I think we can do better in the future. The staff has gained a great deal of experience during the initial reviews and is in the process of compiling an impressive list of lessons learned that can be applied to future renewal reviews. The staff is also developing a streamlined, yet thorough, process for addressing generic issues. I sincerely believe that with additional process improvements, we will be able to complete our review of future renewal applications within 18 months.

In some ways the NRC is a victim of its own success. The Calvert Cliffs and Oconee reviews have demonstrated that the NRC's renewal process is disciplined and timely, and as a result, interest in license renewal within the nuclear industry has dramatically risen. Entergy's Arkansas Nuclear One plant and Southern Company's Hatch plant are poised to submit their renewal applications in the coming months. Many more applications are certain to follow. Sam Collins, the NRC's Director of Nuclear Reactor Regulation was recently quoted in an industry journal as saying that about 85 of the 103 operating units have expressed some interest in license renewal. I don't know what the right number is, but I feel confident that the vast majority of the existing fleet of nuclear reactors will seek license renewal. This, of course, represents a major resource challenge for the NRC. In order to meet this challenge, we will have to capitalize on our experience, get our arms around our process for dispositioning generic issues, and simply dedicate additional resources to our review effort. There is little doubt in my mind that the NRC is up to the challenge and no doubt in my mind that the Commission will provide whatever resources necessary to get the job done.

<u>License Transfers</u>

Now let me discuss license transfers.

As the electric industry proceeded down the road toward deregulation over the last few years, it was clear that the NRC would have to gain a greater understanding of the effects of the changing business environment on plant operations and safety. It was also clear that the corporate restructuring and asset transfers brought on by deregulation would result in an increase in license transfer applications. To ensure that license transfer reviews were conducted in an effective and timely manner, the Commission promulgated regulations establishing an informal and streamlined Subpart M hearing process. In addition, we began work on developing guidance documents to determine whether a proposed transferee is technically and financially qualified, as well as to evaluate foreign ownership and control issues. The overall effect of our efforts has been to improve the preparedness of the NRC, our licensees, and the public for dealing with issues associated with electric utility restructuring.

While the NRC was prepared for an increase in license transfer applications, few of us were prepared for the onslaught of activity witnessed in 1999. We began 1999 confident that the NRC's license transfer process was predictable, disciplined, and prompt. In April, the NRC approved the transfer of the operating license for Three Mile Island Unit 1 from GPU Nuclear to Amergen Energy Company, a company jointly owned by PECO Energy and British Energy. We felt very good because our process proved itself to be sound as the staff was able to address the many foreign ownership issues raised, and still complete its review in less than 4 months. In May, we approved the transfer of the operating license for the Pilgrim Plant from Boston Energy Company to Entergy in about the same period of time.

What was to follow was unprecedented. Announcements were made that ownership of Clinton, Nine Mile Point, Oyster Creek, and Vermont Yankee would change hands. Millstone and Seabrook would be put up for auction. Carolina Power & Light, the licensee for the Brunswick, Shearon Harris, and Robinson nuclear plants would merge with Florida Power Corporation, the licensee for the Crystal River nuclear plant. Then, several weeks ago, two giants in the U.S. nuclear industry, Commonwealth Edison and PECO Energy, announced merger plans. 14 operating nuclear units are involved in this merger. And to think - this is just the beginning.

Clearly, we are in the midst of a truly dramatic shift in ownership of nuclear generating assets in the U.S. Given the current market forces, I believe that this shift will continue to accelerate in the next year or two. As I have said on several occasions, I view the consolidation in the nuclear industry as a tremendous opportunity to further improve the operational performance and safety of these plants. In most of the transactions, I expect that the buyers will be large nuclear generating companies that own and operate a substantial number of nuclear units. These buyers have economies of scale and resources that are simply not available to companies that own and operate only one nuclear unit. I am also truly encouraged by the fact that most of the license transfers will likely involve buyers with excellent performance records. However, as I have cautioned many executives in this industry, these transfers must be managed effectively. License transfers, if not managed properly, can distract management from their primary plant oversight responsibilities and dilute management talent. Since there are typically resource and staffing implications associated with these transfers, they can also distract plant workers from their primary responsibilities. To date, licensees involved in license transfers have done a terrific job managing them. As long as this continues, the Commission is prepared to commit whatever level of resources necessary to ensure future license transfers are handled in an expeditious manner.

NRC Reactor Oversight Program

At the same time that economic challenges in the electric industry are driving the licensees of our nuclear plants to reinvent themselves, government reform efforts and stakeholder dissatisfaction are driving the NRC to reinvent itself. The NRC is now in the midst of one of the most aggressive regulatory reform efforts ever undertaken within the federal government. We are successfully making the transition from big government to good government. In terms of real dollars, our FY 2000 budget has decreased by over 25% and our staffing level is down almost 600 individuals from agency highs in 1993. We are also successfully making our processes more timely and predictable, we are eliminating unnecessary regulatory burden, we are becoming more risk-informed, and we are holding our managers and staff more accountable. Some see these challenges as threats. I see them as tremendous opportunities that I am confident will result in a more effective and efficient NRC.

Never has the NRC had a greater opportunity to reinvent itself as that presented by the new **reactor oversight program**. By almost any standard, the safety performance of the nuclear industry has significantly improved during the 90s and is better now than at any time in the past. The number of initiating events resulting in scrams has declined significantly over the past 10 years, and this is reflected in fewer and less complicated transients. While the NRC's current reactor oversight program has served us well and, I believe, has enhanced reactor safety in the U.S., it has failed to evolve sufficiently as the industry matured and reactor performance significantly improved. Some of our inspection efforts were misdirected on non-risk-significant matters and excessive subjectivity had entered into our plant assessment processes. Quite frankly, our assessment processes lacked the predictability and consistency necessary to run an effective and efficient oversight program.

In response to these shortcomings, the staff is well along its way to developing a new reactor oversight program. The new program will utilize a combination of risk-informed, objective performance indicators and a risk-informed baseline inspection process to measure plant performance. It is designed to focus more of the agency's resources on the relatively small number of plants which experience performance problems, while reducing the regulatory impact on plants that perform well. I'm not going to go into detail about the new program; however, I want to give you a sense for how significant the changes are.

The use of objective and measurable indicators of licensee safety performance should substantially reduce the amount of subjectivity in our reactor oversight process. Initially, there will be 19 performance indicators built around seven safety cornerstones. These cornerstones include such things as initiating events, mitigating systems, and emergency preparedness. I am certain that these indicators and their thresholds for regulatory response will evolve as we become smarter with their use.

In combination with our performance indicators, our new baseline inspection program will be our standard inspection effort that focuses on activities and systems that are truly risk-significant. Additional inspections beyond the baseline will be performed at those plants at which performance is below a specified threshold, based on performance indicators and inspection findings. I am pleased to note that we are capitalizing on our risk research in the development of our baseline inspection program by extracting risk insights from the Individual Plant Examinations to help our inspectors improve the new risk-informed framework. Since these insights focus on areas that drive risk, the inspectors will be able to make more realistic assessments of safety findings, and be better prepared to present their assessments to our licensees and the public.

The NRC is testing the new reactor oversight process with a pilot program at nine nuclear plants. The plants represent a cross-section of the nuclear industry, different plant designs and varying levels of performance. From my perspective, the response to the pilot program has been nothing short of remarkable. The apprehension and resistance to change that is common with most new ideas are diminishing as our inspectors, our licensees, and our stakeholders gain greater confidence in the new oversight program. That is not to say that problems are not being identified. Some believe that our performance indicator thresholds for taking regulatory action are set too high and thus not adequately sensitive to detect performance declines. Others believe the new program could hinder our ability to provide valuable feedback to licensees regarding plant performance. I believe that if the pilot program is effective, more problems will surface. Initially, the pilot program was scheduled to be completed by January 2000. However, given the scope and depth of the changes, the Commission extended the pilots until at least April of 2000 to ensure the new oversight process is adequately put through its paces. I believe this is time well spent.

I am optimistic about the new reactor oversight program. I believe that it has the potential to enhance safety by focusing NRC and licensee resources on the most risk significant aspects of plant operation. It should also bring greater consistency, discipline, and predictability to our inspection, assessment, and enforcement processes. As we proceed along the course we have set for ourselves, we must be vigilant in our efforts to ensure that our new program is able to detect performance declines before they challenge safety. I am confident that the new oversight program will ultimately meet this challenge and I am equally confident that it will increase the level of safety at the plants we oversee.

Research

I'd like to now turn my discussion to a subject that is near and dear to my heart - funding for research projects within the NRC.

As many of you know, the NRC's research program has historically provided a significant part of the Commission's independent technical capability. Congress in the Energy Reorganization Act of 1974 mandated the formation of the Office of Nuclear Regulatory Research to ensure that the Commission would have "an independent capability for developing and analyzing technical information related to reactor safety, safeguards, and environmental protection in support of the licensing and regulatory process."

The scope and emphasis of our research programs have changed over the years as nuclear technology has changed and matured. In recent years, the NRC's regulatory reform initiatives and the nuclear industry's enhanced interest in license renewal, have brought research to the forefront of the agency's efforts. For example, the Office of Research is at the forefront of our efforts to risk-inform Part 50. They are instrumental in developing a new risk-informed inspection program and objective performance indicators. The resurgence of interest in license renewal has highlighted the need for additional research on plant aging, particularly as it affects the integrity of reactor pressure boundary components, vessel internals, electrical components, steam generator tubes, the containment structure, and other passive structures and components. In addition, emerging technologies such as the Framatome electrosleeving technique, bring with them their own unique research challenges.

One would think that with these many challenges, this would be a heyday for the agency's research program. In many ways, it is. However, our research program has never been under greater scrutiny and has never before faced the budgetary pressures it faces today. The days of the open checkbook are over. In fact, the reactor safety research portion of the NRC's budget has declined from over $100M in the early 1990s to around $40M in FY 2000. There are two primary reasons for the increasing budgetary pressures facing not only our research program, but our entire agency - the Omnibus Budget Reconciliation Act of 1990, commonly referred to as OBRA-90 and the Government Performance and Results Act of 1996, or GPRA.

Briefly, OBRA-90 requires the NRC to recover almost 100 percent of its budget by assessing fees to NRC licensees. This has been a highly controversial issue for many years, and one in which the NRC has been actively seeking legislative remedy. With a budget of about $470 million, a declining number of licensees, and increased competition in a deregulated electric industry, our fees represent a significant economic burden to many licensees. It's not surprising that this burden has resulted in greater scrutiny of our budget by our licensees and ultimately Congress. GPRA requires agencies to set long-term strategic goals as well as annual performance goals and to clearly demonstrate how each and every one of the activities they perform is linked to these goals.

Simply put, if you can't demonstrate that what you are doing is linked to your overall strategic goals, you shouldn't be doing it.

The NRC's Office of Research has made progress in reinventing itself to meet the many challenges of OBRA-90 and GPRA. However, one challenge that remains unanswered involves their inability to defend the research program budget to the extent necessary in today's environment. The NRC's FY 2001 research budget proposal, which the Commission recently voted on, focused on describing **what** research projects were necessary. However, one key piece was missing. What was missing was a clear and defensible articulation of **why** these projects were necessary. Missing were the clear links between proposed research activities and the NRC's strategic and performance goals. As a result, I and some of my Commission colleagues aggressively challenged the proposed research budget and quite frankly recommended even further cuts. I can assure you, I will scrutinize the FY 2002 research budget even more vigorously.

I am a fiscal conservative. However, I am not fiscally irresponsible. I understand the meaning of false economics and understand that in order for the NRC to be successful in dealing with such important matters as risk-informed regulation, license renewal, and emerging technologies, our research program must be strong. I also understand the importance of both anticipatory and confirmatory research. Given a sound basis, I will vehemently defend our research budget before Congress, our licensees, and other stakeholders. Without such a basis, I will be research's toughest critic.

I'd like to share with you my perspective on what the budget realities of the new millennium will mean for the future of the NRC's research program. I'm sure many of you who are not from the NRC face similar realities.

First, I believe that the economic pressures facing the NRC and our licensees will manifest themselves into even greater scrutiny of each and every research dollar in the future. If anyone thinks these economic pressures are going away, I urge them to consider the speed at which deregulation is proceeding in the electric industry and the extent to which fiscal conservatism is sweeping our federal government. There should be little doubt that these economic pressures are here to stay. It is imperative that our research staff adapt to this higher standard of fiscal accountability and demonstrate to our stakeholders that our research activities represent an effective use of agency resources.

I also believe that our research staff will have to reinvent the manner in which they defend research activities. Contrary to popular belief, **good research does not speak for itself**. We as an agency have to do a better job articulating why each of our research activities is important to the mission of the agency and demonstrate strong links between each activity and the agency's strategic and performance goals. If the research is defensible, our research staff must learn to market it, sell it, and clearly make the case for why it should be funded. If it is not defensible, the staff should sunset it and move on to higher agency priorities.

I strongly support having a research program that is visionary in its approach and capable of providing independent thought on important agency matters. However, independence has to be carefully managed so that it does not lead to isolation. I believe our research staff must work closely with our program offices and our stakeholders - the primary end users of the research - to ensure that these parties share similar priorities and a consistent, or at the very least a compatible, vision of the future.

I believe that our research staff should adopt a budgeting process that better integrates both micro and macro budgetary components. Too many people inside and outside of the agency are fixated on the bottom line. They point to the bottom line of the research budget and express alarm that it is either too big or too small for the job at hand. I would argue that the bottom line is an almost secondary component of the budget equation. Today's budget realities dictate that we approach our research budget line item by line item. If we are true to our strategic goals and disciplined in our approach, a sound and defensible bottom line will naturally fall out of this process. For those that argue that our research budget is too big or too small, I challenge you to move beyond the bottom line and identify research initiatives that should be done but are not, or research initiatives that are being done but should not.

Finally, while I believe it is prudent for the NRC to maintain a vibrant research program and to assume a leadership role in areas that are critical to reactor safety in the U.S., I believe it is foolish to aspire to be the premier nuclear research agency in all disciplines. The issues are simply too broad and complex for any country to assume such a role. As the regulator of the world's largest civil nuclear program, the NRC has broad capabilities to contribute to international safety and regulatory programs. Also, through cooperation, the NRC often participates in research programs of other countries and obtains valuable information often at a comparatively small cost. For example, International research has already benefitted the NRC in such areas as high burn-up fuel performance and materials issues associated with steam generators and reactor vessel internals. As economic pressures drive greater fiscal restraint, it is going to be imperative that the NRC seek ways to expand on its efforts to capitalize on research work being done by the international nuclear community.

<u>International</u>

While I'm on the subject of international activities, I want to welcome our international colleagues in the audience. It is a pleasure and honor to have you with us.

Your presence here today illustrates how very small and closely knit the world of nuclear power has become. With this closeness comes opportunities. Opportunities to share technical insights and operating experience. Opportunities to exchange views on broad policy and safety matters. Opportunities to capitalize on the tremendous economies of scale of the international nuclear community.

Our closeness also brings with it many challenges. As I have discussed with many executives in the U.S. nuclear power industry, this industry will always be judged by its weakest link. Thus, nation's with mature nuclear programs should recognize the vested interest they have in sharing technical insights, operating experience, and best practices with those nations with struggling programs. We must face the fact that a core damage accident anywhere in the world is like having one in our own backyard. As the accidents at Three Mile Island, Chernobyl, and most recently at the Tokaimura facility in Japan have highlighted, a serious nuclear accident anywhere in the world could have a significant impact on public confidence and could ultimately derail the future of the nuclear option in many countries.

In my opening remarks, I stated that the outlook for nuclear power in the U.S. is brighter today than it has been in a very long time. However, it is not lost on me that the continued safe operation of the existing fleet of nuclear plants around the world remains the foundation upon which the future of this industry will be built. I commit to you today that as the NRC pursues regulatory reform initiatives and faces the challenges presented by electric industry deregulation and government budget cuts, we will not lose sight of our mission to protect public health, safety, and the environment. To do so would simply be irresponsible.

Thank you very much. I would be pleased to use my remaining time this morning to answer any questions you may have.

The Impact of Emerging Technologies on Nuclear Safety Research

John Ahearne, Director
Ethics Program
Sigma XI Center

Let me use my ten minutes to give a little bit of background, identify some challenges, and then at least end with one big problem that I see. As you know, in the announcement for this meeting it mentions the emphasis on research aimed at facilitating risk-informed, performance-based regulation and reducing unnecessary burdens on industry. I will try to get back to those two issues.

Ashok sent a letter in advance of this meeting and he mentioned that the past research work at the NRC provided the essential technical bases for the regulatory framework that has assured a safe nuclear industry in the United States, and has been involved in many international activities. The NRC has worked on such things as replacing crude concepts, assessing the effects of aging, replacing simplistic acts and source terms, and developing probabilistic methods. All of those things are areas in which the NRC research has done work, but clearly has a lot more to do.

Ashok mentioned that the NRC's research program is facing uncertainties and unknowns associated with emerging technologies; and there are a few things that I will try to mention. He mentioned in his letter that adequate time has to be allowed for review by the NRC for emerging technology. The issues that have to be addressed include primarily reactors — this is a meeting on light water reactor safety. I will mention just briefly some of the other issues.

On reactors, in spite of what I hope is an accurate description, a very positive one, from Commissioner Merrifield of the current future of prospects of the industry, I discuss these issues with many utility executives. The sense I get is that many of them are still looking at decisions which are at the margin. That is that they are looking carefully at to whether it makes sense for them to get rid of their plants, sell them at very low costs; close the plants, particularly if they can work with the regulatory — state regulatory agency and do a stranded cost recovery operation — or re-license the plant. I think the tipping factor for those decisions includes NRC actions.

Now, as you know, there's much pressure on the NRC to re-license quickly, without adding substantial cost. It will be critical, obviously, in this recent CSIS Review in which I was involved. Also, Commissioner Merrifield mentioned the NRC has clearly been improving its efficiency. But it's important to also have the knowledge base that the NRC will require. We are well aware of aging questions. The NRC's answers to the questions will depend not only on past research, but to some extent on research either on-going or that has to be done.

Unfortunately, there will always be surprises. For example, the effects on materials. The only way to be prepared is to have a solid research program. One can transfer general technological developments into applications in the reactor world. Clearly, the most obvious, and the one Ashok already mentioned, is in the I&C. As we all know, the computer industry has grown at a pace that perhaps only Moore recognized would happen.

All of us, or many of us now, carry around a small hand calculator that can do as much as a large machine could do a decade or two ago, can do as much as a workstation could do, perhaps a decade ago. The advances are tremendous. The use of microchips has not yet penetrated well into the reactor world, but I think it will.

You may say, "Are there not going to be serious radiation effects?" The Air Force and the weapons labs have spent many years looking at how to harden various electronic circuits, including microchips. What roles could this play in the plant? Well I think in the older plants, which are toward the end of their life, probably not much. For those that are going to be re-licensed, it could be major. That is, the improvement that the microchip revolution could bring could be truly major if it can be shown to significantly reduce the operating cost. Because, in a de-regulated world, the cents per kilowatt hour will be the metric for success or failure.

Are there going to be new plants? Is that a pipe dream? Well, perhaps it is. But, my friend on the right may talk about generation IV plants, which is a vision that the DOE has. The future might be something -- and I will give you a possibility without putting anything on a sense of probability, and then ask what demand this would be on the NRC. Imagine a consortium announcing that they're going to use a completely new reactor design which would include a radically different core and direct conversion for electricity, and that it can beat all the other generation sources for cost, and they'll claim with new, high-strength alloys, they can cut the overnight cost to the range of a gas turbine. Then, finally, they're going to claim that there's a ten-to-the-minus-twelve probability of releases outside the plant in the most severe accident.

So they're going to propose to build this plant within a highly-populated area, for example, within a city. Now they will claim that they have shown extensive, proprietary, materials' tests in Dmitrograd, Russia that the materials' properties meet what their claim is. And they'll also claim, using proprietary codes, that they have developed and run on a multi-teraflop, massively parallel computers at the weapons' labs, to prove this. What does the NRC do?

Currently, with the trend in research, the NRC would probably have to deny the application. Now they may not say they're denying it, they may just say that it's going to take them five-to-ten years to check on the claims. Is it far fetched? Maybe. But surprises come as high temperature super conductors. Those were a bit of a surprise to most people. The manufactured production of elements with atomic numbers greater than 112. Another surprise. How many people would have thought that you could make a molecule with 60 carbons? "Bucky balls." So you can't necessarily conclude that because you're quite confident that you know everything about material properties, or everything about physical behavior, there's nothing new that's going to come along. Let me be more prosaic.

Risk informed performance based regulations — now I'm in favor of the use of risk analysis. I do it and I teach it, but as our CSIS Report said, "Many current PRAs are not done well; and industry and NRC staff needs development — focused development on understanding how to do PRAs, what they mean and how to apply them."

The CSIS noted that to move to transferring all the regulations — major regulations – to risk informed is going to require substantial work, and it must rely on a research foundation that may not even exist yet. That may be the biggest challenge to re-licensing and continued operation.

Another area where NRC research is going to be crucial, or may be crucial, was one that Ashok alluded to, and that's the dual-track approach to getting rid of weapons plutonium. This is a major joint United States-Russian effort. As the weapon stockpile is reduced, HEU and plutonium is released. The HEU is easily dealt with. You can blend it down into LEU for use in regular reactor fuels. To get rid of the weapons plutonium, two approaches have been proposed that are dual tracked. One is to mix it with, or in some way combine it with, radioactive waste and immobilize it in glass; and the other is to make MOX fuel out of it and use it in reactors. But, it's weapons' grade. There's been a lot of MOX fuel used in Belgium, France and Germany, but not using weapons' grade plutonium. Does it make any difference? Well, let me just refer to a few things from a recent paper that points out that the primary difference, of course, between reactor grade plutonium and weapons' grade plutonium is the isotopic composition; but in addition, weapons' grade MOX will have traces of gallium that can react with zirconium alloy cladding.

The plutonium reduces the reactivity worth of control blades, burnable poisons, and liquid absorbers that makes it perhaps more difficult to control the reactor. There is a lower delayed neutron fraction, which also makes reactor control more difficult; and a harder spectrum so it can increase the fluence on the vessel.

MOX fuel pellets operate at a somewhat higher temperature during normal operation. It can affect initial conditions for loss of coolant accidents and increase gas released during normal operation. The reduction of the effectiveness of the reactor control system is perhaps the most important technical issue that must be addressed in the plutonium disposition program. The paper points out, the NRC analytic codes will have to be upgraded to include MOX-specific parameters. The NRC and industry codes will have to be modified to model MOX fuel adequately, and then data is going to have to be obtained, perhaps from the Halden Test Reactor in Norway.

You may say, "Well, MOX fuel's been used in Europe. Reactor grade plutonium can't be much different." For those of you who have been involved with any of the licensing controversies and environmental issue controversies in the United States, you have to recognize that major challenges will be raised against the use of MOX fuel.

There's very strong opposition to that part of the dual track. The NRC's research foundations will be critical to whether the United States can move forward down that path. Not unrelated to the existence of light water reactors continuing will be waste management issues, including safety at the sites. In my state of North Carolina, there's now ongoing the process, which I think is going to lead to hearings, where Carolina Power and Light has attempted to consolidate spent fuel into one of its reactors that had an unused spent fuel pool. There's a big challenge on whether this is going to make an unsafe situation, and then, of course, all the issues on transport. Accidents happen. The recent Japanese criticality accident was caused by human error, although error may be too mild a term. Humans are the weak link, or they can be the last recourse.

Let me close with first just a little anecdote, and then the significance of it. Sherlock Holmes and his colleague Watson went out and camped out. They pitched their tent and they went to sleep. Holmes woke up and nudged Watson and said, "Watson, tell me what you see?" Watson looked and said, "It's a beautiful, crystal clear night and I can identify the constellations, and I can see by the constellations where approximately we are, and I can also understand some of the heavenly movements." Holmes said, "Watson, you fool, somebody stole our tent." Sometimes the obvious is missed.

I have a major concern. There apparently is a belief in elements of Congress, the administration, industry and perhaps even in the NRC, that research is not needed. One can see this in other industries as R&D labs are drastically downsized or closed. Vannever Bush many years ago — 50 years ago — wrote a critical document calling "Science: The Endless Frontier," and he tried very strongly to point out that basic research has to be done without the need to knowing where it's going to lead. That's what research is.

Donald Stokes a few years ago wrote a book called "Pasteur's Quadrant." It pointed out that mission-based research, use-inspired research still has to have that essential quality of research. You don't know where it's going to end up.

NRC's research budget has shrunk to a very low level. It perhaps may soon approach the point that when a new issue comes up, the NRC will have to say, "The issue can't be addressed." Perhaps this has been a stealth tactic to shut down all nuclear power plants.

The Impact of Emerging Technologies on Nuclear Safety Research

Joe F. Colvin
President and Chief Executive Officer
Nuclear Energy Institute

Thank you for the opportunity to be here today and to be part of this distinguished panel.

These are vital matters we are addressing today. The very nuclear energy industry we're here to advance has its roots in the tireless innovation of diligent researchers . . . those visionaries who first crafted the government/industry partnership more than four decades ago. It is they who had the vision and passion to innovate, to study and to craft the commercial nuclear energy industry we know today.

And what incredible results . . . an energy technology that now produces 20 percent of the nation's electricity—emission-free . . . a vibrant industry with the ability to thrive in a newly competitive electric power market . . . a source of power with tremendous potential as a clean air compliance tool.

And the exciting thing is that we have a remarkable opportunity here today to help shape, refine and rejuvenate that spirit of innovation. And the timing couldn't be better.

Thanks to the commission's leadership, the NRC is now engaged in an unprecedented shift in its approach to regulatory oversight, focused on plant safety as never before.

The agency is shaping a new strategic plan that proposes a "broader context for success" and that moves the agency from an "*output*-based environment to an *outcome*-based environment."

What better time to revisit the fundamental tenets of NRC research? To assess its likely outcomes . . . its value to a rapidly changing industry . . . and its focus on safety.

The U.S. nuclear industry today is performing better than ever before. Last year, the average capacity factor of our nuclear power plants reached a record high of 79.5 percent.

This year, we're well on the way to setting a production record: During the first half of this year, our 103 nuclear generating units produced 347 billion kilowatt-hours of electricity—up 9.5 percent from 1998.

For the last 12 years, our production costs have been trending downward. And I'm proud to note that this excellent performance is being accomplished with a remarkable record of safety—both for our workers and the public.

This concrete record of outstanding performance is not happenstance. It is the result of two decades of continual improvement . . . innovation . . . and *applied* research and development.

Never has the time been so promising to forge new government/industry partnerships . . . to research and actually implement emerging technologies!

There is a clear precedent for such joint ventures . . . the very successful program to develop Advanced Light Water Reactors, jointly sponsored by the industry and the Department of Energy earlier this decade.

And we've just heard Bill Magwood describe another promising joint program . . . DOE's Nuclear Energy Plant Optimization program.

These DOE programs could well be models for programs proposed within the NRC's five-year strategic plan.

They are based on practical outcomes that will improve both the performance and the safety of nuclear power plants in the United States. And they set the stage for the future, when we will build a new generation of advanced nuclear power plants.

Let's take a look at what made the ALWR program successful.

First, it had clearly defined objectives. Both industry and government knew where we were headed and what we wanted to accomplish. The program was outcome-based . . . not output-based.

Second, the program involved organizations from all parts of the industry: NEI . . . reactor manufacturers . . . utilities . . . international organizations . . . special advisory groups . . . EPRI . . . INPO . . . NRC . . . and DOE.

Third, at every turn, the participants' work products were defined, and they knew how their outcome-based results were to be integrated.

Fourth, we maintained tight fiscal control and monitoring. We brought good management practices to bear as we worked through various phases of the ALWR program . . . because we knew the program was clearly an investment in the industry's future.

By contrast, let me mention three examples of NRC research efforts, which I believe have not followed these principles and, as a result, have not been successful.

First, the NRC's research on reactor coolant pump seals . . . a project that has gone on for almost 20 years with no concrete results.

Second, the development of a standard for probabilistic safety assessments . . . where one NRC office has issued guidance to licensees regarding an acceptable standard and another NRC office has contracted with an independent standard-setting agency, the American Society of Mechanical Engineers, to develop a separate standard.

And third, the tendency of the NRC to engage in confirmatory research that duplicates industry efforts. We have a great opportunity to refocus our efforts.

I can think of no better time to eliminate NRC research efforts that are duplicated by licensees and other organizations.

I can think of no better time to phase out longstanding, ongoing research that has produced few results.

And I can think of no better time to reassess the value of every research dollar spent . . . and to build the research base for continued and improved plant operation.

The real opportunity I see is for the NRC and industry to craft common goals for key research, so that the agency and industry can jointly plan, manage and finance the right projects.

These projects should meet preset standards that factor in the appropriate level of research to enhance the industry and NRC needs . . . that require research that is focused on matters of safety . . . and that enhance plant performance, now and through their extended license periods and, most importantly, will lead to actual use of new and emerging technologies at our plants.

Finally, the NRC's research projects should not reinvent what the industry or other organizations are already doing. We simply cannot afford to waste our limited resources in duplicative and confirmatory work.

In its fiscal 2000 budget and performance plan, the NRC set a goal of issuing 24 research reports that range from resolving generic safety issues to serving as the basis for agency decision making.

Instead of setting a target number of research reports, I would recommend that the agency evaluate its year 2000 goals based on desired and expected outcomes . . . eliminating confirmatory research of work done elsewhere . . . and undertaking only those projects that focus on issues of safety significance to plant operations.

It is my opinion that confirmatory research is simply not necessary. In cases where the industry conducts research that adequately addresses a public health and safety concern, the NRC could instead follow up by reviewing and using industry research results . . . not by starting all over with redundant research.

We have seen the beginnings of change already. I am encouraged by the NRC's attention to validating changes initiated by plant licensees themselves.

The Office of Nuclear Regulatory Research has, for example, supported recent licensing actions on new steam generator repair techniques at the Callaway plant and new ways to monitor feedwater flow at Comanche Peak.

This kind of activity holds promise to usher in new dialogue regarding industry strategies to marshal new technologies.

The underlying and longstanding benefit I see ahead is a research effort more focused on the one thing that matters most to us all . . . plant safety. Just as a new safety-focused regulatory paradigm has led to a sea change in reactor oversight, so too will a renewed attention to safety-focused research reinvigorate the agency's research programs.

The need for increasing amounts of electricity in the United States shows no signs of abating in the decades ahead.

The continued use of baseload-capable, emission-free nuclear power plants in meeting those needs is essential. So, too, is the continued development and application of emerging technologies.

I have confidence that industry and government can continue to forge the alliances needed to produce the kind of research that truly benefits our nation and the citizens we serve.

Thank you.

The Impact of Emerging Technologies on Nuclear Safety Research

Jim Lang, Director of Power Production
Electric Power Research Institute

I don't think we have any choice but to deal with emerging technologies. If you take a look at our industry, to a very large extent, the technology was codified and frozen in time decades ago. Analog to digital conversions are a great example of that. We've had a challenge and have been quite slow to accept the challenge of moving on. We do have the advantage there that other industries have provided pioneering in this area decades ago, so maybe the challenge shouldn't be so great.

We're also going to be confronted onward, as we move onward with technologies that have the promise of reducing costs, but I have to say, technologies in and of themselves hardly ever reduce cost. In fact, if organizations and management aren't in place to use them, technologies tend to increase cost.

Finally, some technologies produce uncertainties, and I think that's a very important thing to acknowledge here. That's the case in the vessel embrittlement area. That's the case in fatigue, where we establish design rules with great margins because of uncertainties. Now with research results, we're able to evaluate current installations with much greater precision and accuracy.

We really shouldn't be afraid of technology. Technology is our friend. I mean, we don't have any choice but to apply much of it. As I said before, time marches on. You can't buy analog equipment anymore. We really need to adjust. But as we proceed, we need to assess the risks and prioritize the concerns — and that's true on the industry side and that's true on the regulatory side.

If you take a look at the industry research program right now in nuclear, it's pretty stable at about $90 million a year. That's a voluntary program. Utilities ante up that kind of money on the nuclear side, not because they're coerced into it, but because they believe there's value there.

Much of the value is driven, obviously, by economic considerations, but let's not lose sight of the greatest economic challenge to any nuclear operator today, the threat of an accident. So, virtually every aspect of our research or every piece of our research contains some safety aspect. It's part of the culture, and it's necessary to assure a sound future.

As we move on and evaluate new technology and the implementation of new technology, I think it's really important for us to focus on how to practically apply it, rather than all the reasons we shouldn't apply it, which is the trap we fall into.

Commissioner Merrifield referred to our risk aversion a little bit earlier when he presented the risk award dilemma.

Finally, as we march ahead, we have to maintain our technical integrity. Now that's obviously motherhood. We can't select the data that we like. We shouldn't selectively apply results, and absolutely none of us in this room would condone that in justification of the application of new technology, yet we do it all the time in the name of conservatism.

We, in looking at technology, have to define both technology and research broadly. There's a certain snobbishness sometimes in research, and I think I sensed it in some of the deliberations of the NEPO Committee a few weeks ago, that research is what takes place in the laboratory and the rest is all a lesser endeavor.

But the implementation guidance has to be developed, and we have to be concerned about worker response as they move forward to implement. We have to pay as much attention to the back end as we do to the front end.

The NRC and the industry, in looking at research regarding emerging technology, really have complementary roles in my view. Those who want to apply new technology need to provide the technical basis. But the NRC certainly has an obligation to be informed. The health and safety of the public demands that, and the industry's interest demands that.

The old saw "the knowledgeable regulator is a good regulator" is not just an old saw. I think it's true. But there's no need — and we go back to everybody else's theme — there's really no need to duplicate the research.

There is a need for the NRC and industry to coordinate their approach. Early and frequent communication and cooperation is required at the front end. Joe presented the example of the ALWR Development as an example of that. Bill Magwood emphasized the importance of that in his discussion.

Research can be jointly defined. It isn't easy because there are competitive views here. But research can be jointly defined and conducted to gather the needed data. If we meet early enough, we define the needs early enough and we plan together.

The data can then be interpreted independently, and I think independence is important. It's important for public perception, and it's important for the NRC to fulfill their duty. I don't think our goals and objectives are all that much different, but there is a kind of seedy perception of us in bed together.

It's been mentioned before that NRC research and EPRI have a memorandum of understanding, and that memorandum outlines a process very much like all the people on this panel have promoted and that I've just described. The challenge for us now is to take something that sounds good, and I think that we would all argue that it sounds good, and make it work — and that's much more difficult.

It did work in ALWR. We have made it work in license renewal and life-cycle management in the past, and we're marching down the path to make it work now. Together, the NRC, EPRI and DOE are working on characterizing spent fuel that's been in dry storage for about 14 years. The idea is making sure that the standards that we have are valid, and that we know where we're going in the next ten years.

Risk informed fire protection data and models are being prepared jointly by EPRI and DOE. Welding of reactor vessel internals — weld repair of reactor vessel internals is an issue that can be affected by the very high irradiation of the materials. NRC and EPRI are working together right now on welding of highly irradiated materials.

Again, there's an input there from the Japanese — an international element — where we can realize a much more economical approach by borrowing from people who've been there before.

High burn up fuel issues is another area where we have a very good track record. The French have done a lot of research on high burn up fuel, and we had access to literally millions of dollars of French results by virtue of cooperative programs between the industry and EDF.

The NRC also had access to that through the French regulators, and we're working together. There are still a lot of issues about where do we go on high burn up fuel, what tests are necessary, and so on. But if we work together, we're going to get there and we're going to get there more economically.

Those are my remarks.

The Impact of Emerging Technologies on Nuclear Safety Research

Bill Magwood, Director
Office of Nuclear Science and Technology
U.S. Department of Energy

Thank you Ashok. I should say that there is a mixed blessing in following John Ahearne, but I guess it's better than coming before him, I don't know. I guess it could go one way or the other. I agree with a lot of the things that John had to say, but I think there is another dimension of this I'd like to get into.

First, it's worth pointing out that this year, for the first time in probably more than a decade, the DOE Nuclear Energy R&D budget is actually larger than the NRC R&D budget. This is the first year that's happened. I think we're up around $50 million and NRC's around $40 million.

I think that John's points were all very well taken, that the research needs to get done. There are a lot of important issues out there that need to be investigated. Materials issues need to be investigated, technology issues, digital I&C, all these things need to be done.

I think that one of the key questions we have to think about is, what NRC activities really should be in relationship to these R&D activities. About three or four years ago, NRC and DOE worked together on a program to develop — well, let me rephrase it - we worked together, but it wasn't working together to develop technology.

It was a DOE program to develop technologies to anneal reactor pressure vessels. NRC joined us in the program in a manner in which they were not working on technology development. What they were doing was observing and taking their own data as the process went forward.

I think that sort of program is probably -- when you're thinking about the emerging technologies -- the sort of activity that NRC ought to become more engaged in. Rather than have independent nuclear research and I think the word "independent" is a word that we have to think a lot about. Rather than have independent NRC programs where NRC is going off doing its own research in an area, it makes a lot more sense for NRC to work with the industry and work with DOE and maybe others to observe and take data and have another type of relationship as technology programs go forward.

There are a lot of good opportunities to do that, especially with these new emerging technologies. In our program, we're starting a program called "Nuclear Energy Plant Optimization Program," which is a joint program with EPRI that's now just getting underway, and there are a lot of items on the agenda for that program.

We have already invited — and I think NRC is not working with us — invited NRC to work with us in that program to be an observer, to participate in some of the discussions about which research should actually be pursued and what the priorities are. I think that as this program goes forward it makes perfect sense for NRC to become engaged in that program, directly. To observe what's happening. To have their own people, their own resources, observing the results of the research, collecting the results of the research and understanding it. I think everyone understands that materials' issues will be very important for both the present plants and future plants. There's no reason for that data to be taken twice. There is no reason for experiments to be done twice. I think it makes more sense for NRC to be there, to collect the data that we collect, and to then go off and do their own analysis and come up with their own independent conclusions. That of course takes resources, but takes less resources than doing its own experimentation.

I think in addition that there will also be an opportunity for NRC to engage with DOE and others in advance technologies. John mentioned our discussions about Generation IV nuclear power plants. I think Generation IV probably looks something like what John was sketching out when he was talking about sort of a theoretical, future reactor system that would have direct conversion with a 10-to12th probability for releasing materials. That type of system will clearly require significant NRC interaction to make sure that if it ever does appear on the screen that we don't spend ten years trying to get it licensed. We need to be certain that the NRC understands how to license it.

Really, I don't think it's a very farfetched example because we have an example already in our past where we went into this; and that was the Westinghouse AP 600. The AP 600 was a program where we thought in working with NRC up front that we could get this certified, basically, on the same schedule as we got the advanced spoiling water reactor and system eight plus certified.

Well, as you all know, the Westinghouse system still isn't certified, and our participation in the program has ended. Now a large part of that was that we had to spend — I think it was about $30 million extra dollars and two or three extra years — to do a lot of verification work for NRC's benefit that no one ever thought we were going to have to do.

Now, I'm not pointing fingers at NRC. It's just the way that things worked out. But I think that that shows the potential for there being huge delays in licensing new technologies, because of the fact that NRC doesn't have the research results and the background. So, what we think is important is to bring in NRC along every step of the way in these programs, and to have the work directly with the leader, work directly with the industry. I don't think the NRC's independence is impacted by that if they're doing the analyses themselves.

Another technology that we're going to be thinking about, and I know all of you are aware of this, is accelerator transportation of waste. There is a new program that has started in DOE to look at ATW. Well, how are you going to license sub-critical reactors? That's going to be a major question. How do you license lead bismuth coolant systems?

You know, the NRC has no experience in these areas; and if they're going to be able to license these systems, if they ever do come to the point of being licensed, it means that they'll have to be involved from now until the time we actually develop the systems.

They need to understand the same systems, the same materials, the same processes that we need to understand in order to be able to regulate safety. It makes absolutely no sense for us to develop these technologies and show up on NRC's doorstep 20 years from now and say, "Okay, license this." It would be absolutely irrational for us to do it that way. So we need to bring NRC along. NRC needs to become more integrated in what we're doing, and I think they need to find ways of maintaining their independence while yet being involved in these programs.

I think that there are lots of opportunities over the next several years to bring NRC along on programs like that. I think that the industry will be willing to do it. I know that EPRI and NRC already have an understanding related to high-burn-up fuel, for example. As Ashok mentioned, there's a memorandum of understanding between DOE and NRC. So I think that the opportunities are there.

I think that there's a lot of willingness to work together; and I think quite frankly that if NRC is willing to basically throw a lot of the research burden back on DOE — not all of it, but a lot of it — it will also make it easier for us to make the case to Congress that, first, we're not in conflict with what NRC is doing and we're not duplicating what NRC is doing.

Secondly, we will be able to show that NRC sees value in a lot of the work that we're pursuing and wants to be involved. Precisely how that interaction, I'll leave up to the politicians, but it may make sense some day for DOE to have enough resources to simply have NRC participate with us and eliminate the cost to NRC. I think that may be a more rational way of having NRC's research funded in the future, as opposed to having it funded by the rate payers.

Finally, I think it's important to recognize that there will always be some activities that NRC is not going to want to share with the industry, because of their independent role; and that will almost invariably focus on existing nuclear power plants. That, I think, is a clear role the NRC will want to maintain into the future; and I think that's appropriate. But I think that over the longer term that maybe this is more of a prediction of the future than simply an indication of what I think should happen, but I think that if you're really going to have the NRC involved in these emerging technologies, it will have to be as a partner with DOE and the industry, as opposed to carrying out an independent research program.

Thank you.

OPENING STATEMENT ON EMERGING TECHNOLOGIES

by

Ashok C. Thadani, Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

The Kyoto Protocol and concerns about global warming have provided an opportunity for the nuclear industry to emphasize the advantages of nuclear power. At a different level, deregulation of the electric power industry and renewed emphasis on non-proliferation are presenting new challenges regarding the role and future use of nuclear technology. Addressing all these challenges will require that the nuclear industry consider emerging technologies and improved designs, if nuclear power is to play an important role in future generation of electric power.

Deregulation is placing new demands for nuclear plants to operate more efficiently, so they can compete in a economically stressed environment. Federal Energy Regulatory Commission Order 888, issued in 1996, requires that utility and non-utility generators have open access to the electric power transmission systems. Production is being separated from transmission and distribution, affecting plant ownership, and financial structure of utilities. Utilities are under pressure to reduce costs, and find ways to eliminate unnecessary operating burden. Nuclear power plants must now compete with all forms of electric power generation, and this provides a strong incentive for a nuclear enterprise to employ today's emerging technologies.

The technical basis for implementing an emerging technology is, however, frequently limited, so that independent, confirmatory research is often required to close remaining issues. For example, an enterprise could be preoccupied with the new technology's capabilities and benefits, and overlook the risks associated with an emerging technology. An example of such an issue, which I will discuss a later, involves beyond design basis performance of steam generator tubes following an electrosleeving process. Consequently, independent research to identify, analyze, test, and assess operating experience to better understand the safety implication of new technology, remains an important part of research today. These activities naturally evolved from those independent research activities that set the stage for emerging technology. Advancements in probabilistic risk assessment techniques, for example, provided the framework and tools for risk-informed regulations. Source term research established new regulatory criteria on containment performance, and aging research is being used today to support license renewal.

Early involvement by the regulatory agency is the key to sound technical decisions. Let me discuss a few examples of emerging technologies currently under consideration by the industry. The examples fall into two categories, the first involves emerging technologies that are intended to reduce operational costs, the second involve advancement in state-of-the-art technology.

The operating costs of the nuclear industry can be reduced by increasing cycle length and extending the life of critical components. Increased cycle length reduces fuel costs and the need to purchase power during nuclear plant shutdown or operate older less efficient fossil plants that are maintained as standby units. Also, longer cycle lengths result in fewer spent fuel elements and reduced requirements for on-site storage capacity and ultimate fuel disposal. Cycle lengths have increased considerably and maximum fuel burnup is now about 65 GWD/ton. New fuel and cladding designs under development, could result in a further significant increase in the burnup limit. With research engaged early, NRC safety decisions will become more effective, efficient, timely, and preclude the need for regulatory conservatism to address uncertainties that result from lack of knowledge. However, long lead times are needed to perform the necessary experiments to demonstrate fuel and cladding performance under accident conditions.

A number of pressurized water reactors have also incurred significant costs related to the replacement of their steam generators. A new process, called electrosleeving, has been developed to repair degraded steam generator tubes. This new process deposits an almost pure nickel sleeve inside an existing steam generator tube to provide a structural replacement for the pressure boundary in that area. RES and the Argonne National Laboratory conducted a critical review of the corrosion performance of the electrosleeve material and concluded that its performance was substantially superior to Inconel on the basis of design basis conditions. Under beyond DBA conditions, however, uncertainties remain. We were fortunate that the Argonne facility was available to assist the staff in this evaluation. Research activities that address technical and risk important issues associated with the electrosleeve process will support the NRC performance goal of maintaining safety.

State-of-the-art technology and potential for cost reduction is also moving licensees from analog to digital instrumentation and control (I&C) systems. These systems provide various monitoring, control and protection functions. Digital systems have important advantages over existing analog systems, including higher degree of accuracy, high data handling capabilities, self-testing, and process system diagnostics. In addition, digital systems have lower maintenance costs and fewer inadvertent shutdowns and challenges to safety systems. These advantages combined with the potential to monitor equipment on-line using condition sensors, can make digital I&C very attractive. Nevertheless, not all licensees have moved to digital I&C systems because of remaining uncertainties in system performance in degrading environments, and software bugs that could compromise the system performance. Current research programs that build on Advanced Light Water Reactor Program reviews (which utilized state-of-the-art digital I&C system in seeking NRC approval), address these uncertainties and provide the means to support risk-informed regulatory decisions.

And as a final example of new emerging issue that places RES at the forefront of new technology, stems from the nuclear industry's renewed interest in non-proliferation programs resulting from the large surplus of weapons grade plutonium. One proposed method for plutonium disposal is to fabricate the material into mixed-oxide (MOX) fuel rods for irradiation in commercial reactors. The goal of this program is to make the plutonium inaccessible and unattractive for weapon application. It is anticipated that the irradiation campaign could use 35-40 per cent MOX fuel core load fractions and cover a period of 12 years starting around 2007. RES plans to test and develop a data base to address MOX-related technical issues, which will lead to near future sound regulatory decisions.

Regulatory research to independently confirm the safety of these and other new technologies must be planned and conducted sufficiently in advance to ensure that regulatory approvals do not delay acceptance of these technologies. For example, review of emerging technologies often requires the development of improved computer programs for assessment of postulated accidents and operating transients. Extended time is often required for generation of experimental data needed to develop new models and for validation of the computer programs. Examples of such code development include the new source term characterization, determination of the residual life of components and structures, and analysis tools for probabilistic risk analyses of reactor system behavior. Consequently, it is important for industry and DOE to come forward to the Office of Nuclear Regulatory Research and identify their future needs so that the NRC can make appropriate plans. Such action will enable the NRC to make timely, robust decisions and improve public confidence in nuclear power Otherwise, the NRC will become a cause for delay in the use of new concepts and technology.

Identification of future needs is also essential to help ensure that sufficient research capabilities (expertise and facilities) are maintained in light of shrinking resources. This continues to be a problem facing the research community and has long range implications for operating as well as future plants. Identification of minimum research needs is a world-wide issue needing international attention and solutions. Such attention is the focal point of a study currently underway in the NEA/CSNI. In a draft CSNI report on nuclear safety research in OECD countries, concerns have been raised that dwindling budgets and support as well as stagnant programs may lead to untimely shutdown of large facilities and the breakup of experienced research teams. The consequence of such losses will be reflected in the competence and capability necessary to deal quickly and efficiently with future safety problems. The report indicates that specific action is now justified in order to address these concerns which is perceived as serious for the Industry and Regulators, and which is getting worse. These actions include, for example, identification of potential facilities for present or future international co-operation, joint research projects, shared data banks, exchange or sharing of experts, and joint development of computer codes.

We are looking forward to the final CSNI report, but already the need is clear, loss of critical competencies and important experimental facilities will threaten our ability to regulate and provide the checks and balances that ensue safe plant operation. We can all agree that International collaboration is essential, to sustain key expertise and facilities throughout the world, necessary to ensure the safe generation of nuclear power.

Implementation of Emerging Technologies in Nuclear Reactors: A Necessity

by

Alain Vallee, Vice President
Corporate Research & Quality
Framatome

The first big challenge we have to face is to improve competitiveness of the nuclear reactors.

The pressure to get costs down is strong due to the drastic improvement in competitiveness of the conventional plants. The fuel prices (gas and coal) are now very low and during the last 10 years the plants efficiency have tremendously increased. Even if the fuel prices should be back up within the next 10 years, there is no hope to see the nuclear ranking first again without any effort because the trend on efficiency in thermal power plants will continue.

On the other hand, we have to keep in mind that in this economical competition with the conventional plants, nuclear energy has a specific handicap due to the fact that R&D costs are intrinsically much higher, due to the cost of the neutron flux.

Some other factors will influence these next 10 years:

- The deregulation movement, already in progress, will continue, with the progressive disappearance of protected national markets, decentralization of electricity production means to get closer to customers and the appearance of new private electricity suppliers, with more limited financial resources, the necessity of a short term return of investment and greater exposure to the risks of competition.

- The extension of the notion of the offer, with more comprehensive packages, associating the supply of electricity with the products and services.

- Industrialization of a whole subcontinent like China, but also countries whose needs and electric power grids are more limited at the outset.

So, as the market changes, the products will have to evolve, and the reactors to be ordered in 2010 will differ significantly from those of the 70's.

The second big challenge is evidently the necessity to improve environmental perception of nuclear energy by the public.

One condition stands on : the safety aspect has been and will remain essential. The consequences of a core melt accident being unacceptable, the probability of such an accident must be reduced to extremely low levels.

As concerns nuclear energy, public opinion is also very sensitive to the still incomplete solution to the problem of disposal of high level radio active wastes. This question goes far beyond the framework of new reactor design, but new models must at least minimize the activities of such wastes production, and their potential capability to burn stocks of civilian or military origin will be a non negligible advantage.

<u>How to implement new technologies without putting on them an unaffordable burden either in delay or in cost</u>

The problem of implementing new technologies in the nuclear reactors, while keeping a high level of safety is not a recent one; it is as old as the nuclear energy and it was mastered successfully in the past . In this room, there are many safety experts, and any of them can correct me if I am wrong, but I have the feeling that there was no more accidents on nuclear reactors at the beginning of civil nuclear energy than now. So, there is no reason to fail in the future. Moreover, we now have a large experience in the way to handle the problem and may be we are able to derive some basic rules from this experience.

New technologies can be implemented in different areas of activities in nuclear reactors, typically:

- Design
- Construction and manufacturing
- Operation
- Maintenance and repairs

In order to try to define some general rules, I shall give you some experience we gained on the subject from EPR development, which came from a design phase, and from our repair activities, coming from US experience.

For the EPR development, our main targets were to take profit from our past experience on LWR, the German one and the French one, while incorporating new technologies for improving the level of safety. This improvement was expressed by two main objectives:

- A probabilistic one: the achievement of a global core melt frequency of less that 10 –5 per plant operating year, uncertainties and all type of failures and hazards being taken into account;

- A deterministic one: despite their low probability, accident situations with core melt which would lead to large releases have to be practically eliminated. Core melt sequences will require only very limited protective measures in area and in time for public (no need for emergency evacuation outside the immediate vicinity of the plant, limited sheltering, no long term restriction in consumption of food).

To meet these objectives, the support of a large R&D program was necessary with the implementation of new technologies. The interaction between scientists, engineers and safety experts was organized to design the reactor in a way which meets all the requests from the Safety Authorities.

Let's analyze more deeply two technologies dealing with the severe accident approach:

- The loss of the cooling of the fuel is postulated in the course of some scenarios leading to reactor vessel melt through; so the corium is spread outside the vessel and has to be maintained and cooled inside the containment. The spreading and the cooling take place in a specific room where the concrete has to be protected by a layer to keep its integrity at the temperature of the corium. The zirconium oxide was chosen as the material for the layer, and tests were necessary to prove the efficiency. Some tests, such as in the Phebus facility of IPSN, showed that the corium react with zircon to create an eutectic which melt at 2300 °C instead of 2700 °C as expected. The result was a completely new approach for the design with implementation of a specific concrete gate, reducing the temperature of the corium to an acceptable temperature.

- Another technology was developed to reinforce the third barrier for providing supplementary margin: a composite layer is fixed on the sensitive inner parts of the containment to avoid leaks through the concrete. The development to find a material able to sustain all the loads in the course of a severe accident was long and the final implementation in the project is recent.

So, we can conclude that a large R&D effort, with extensive qualifications, is always necessary before implementing new technologies dealing with safety in a new design. Such programs must be launched at an early stage of the design and they can be on the critical path of the design phases.

<u>Interference between new technologies and safety rules</u>

The two previous examples from the EPR are related to the development of new technologies in order to answer the request from Safety Authorities for an improvement on the level of Safety of the future plants compared to the current ones.

Then, two new categories were added to the classical classification of accidents:

- The first one is derived from the Probabilistic Safety Assessments performed on LWR and takes into consideration accidents scenarios with multiple failure and common mode failure.

- The second one deals with severe accidents assuming a consecutive vessel failure. In that case the scenarios were chosen on a dual approach deterministic – probabilistic, based mostly on the scientific and engineering judgements of experts.

As we are entering in a new field in nuclear safety, dealing with events of very low probability of occurrence, with insufficient reliability data and physical phenomena not fully understood and described, the iterative process between designers and safety experts, as done in the first phase of development of nuclear energy, must prevail.

But let us have another specific example, not coming from the EPR but picked out from our recent US experience.

The electrosleeving of steam generators on site is a way to extend plant life at a cost significantly lower than steam generators replacement. It consist of a nano - crystalline nickel sleeve that is electrochemically deposited on the inner surface of a steam generator tube to span a known flaw. This is a very good example of a new technology having the capacity to improve competitiveness of operating plants. During the licensing process, the staff raised concern regarding the performance of the Electrosleeving during beyond design basis severe accident conditions : the Electrosleeve material would weaken during high-temperature severe accident scenarios as a result of grain growth. So, this technical issue highlighted the need for clear policy and process guidance on how to deal with proposed license amendments that are not risk-informed, satisfy existing design and licensing base, but introduce new potential risks.

Conclusions

The first conclusion is that incorporating new technologies in nuclear reactors is an already existing but long process.

One key for the future of nuclear energy is to ensure the public acceptance, so no one can envisage or justify a reduction of the necessary effort before implementing a new technology.

However, it's important for industry not to have long and uncertain processes, otherwise it can kill the creativity and consequently kill also the nuclear energy.

Maybe, some guidances have to be defined in order to make the licensing process efficient that at each stage of the development the designer is able to master its own risks. Such guidances have also to take consideration of the necessity to keep the independence of the Safety experts.

The second conclusion is that improving scientific knowledge is beneficial both for safety and competitiveness:

- It can help to define accurately where the risks are
- It can relax undue margin taken as provision

So, safety of reactors and nuclear development are closely linked if the improvements in the safety are recognized and improve the public perception.

But this is another story...

NEW APPROACH TO THE WRSM 99 MEETING FORMAT

by

Margaret V. Federline, Deputy Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Good Morning! My name is Margaret Federline. I am Deputy Director of the Office of Nuclear Regulatory Research. Welcome to this the 27th Annual Water Reactor Safety Meeting (WRSM 27)

This is a time of fundamental change for the NRC and the Water Reactor Safety Meeting reflects these changes. I want to share with you the thinking that went into the changes in format and content that you will see in the meeting over the next several days. In the Office of Research, as for the whole agency, our fundamental mission is the protection of public health and safety. But in judging our success as a regulatory agency, we must look more broadly and consider the impacts of our work on the public we protect, the industry we regulate and the efficiency, effectiveness and realism of our own activities. RES is no different. We are in the process of moving to an outcome based organization and defining our success in terms of these outcomes. We recognize that RES must be responsive to the changing environment and the implications of a maturing nuclear industry. The nuclear industry is undergoing economic deregulation and restructuring which will increase pressures to use risk insights to gain efficiencies. Through our performance measures we must recognize the need for tools, methods and standards to make a smooth transition to a risk-informed regulatory process. With an aging industry comes the possibility of new safety issues; but also the benefits of research and operating experience to introduce realism in the regulatory process. RES must contribute to these outcomes. If industry is to develop and take advantage of new technology from other industries, RES must anticipate these developments and lay the technical groundwork acceptance of these technologies. The WRSM has been designed this year to illustrate the relationship of RES activities to meeting agency goals. I want to share with you our thinking behind the design of the meeting and ask for your feedback on its success as the meeting progresses.

As a centerpiece for our meeting, we are fortunate to have a number of recognized experts serving on two plenary panels to address the role of regulatory research in facilitating the development and implementation of new technology and the research role in balancing maintaining safety and reducing unnecessary burden. Those of you who have participated in our agency strategic planning activities recognize the link between these topics and agency performance goals.

Our objective for the WRSM meeting is to promote serious technical dialogue on these and other related issues which have the potential to impact key regulatory outcomes. All of the sessions flow from two themes which capture how we are judging our own success: Regulatory Research to Facilitate Risk-Informed Regulation and Reduction of Unnecessary Burden to the Industry and Research to Respond to Emerging Issues and Operating Experience. Our co-chairs and presenters have been invited to represent the range of experience and technical views in each of these areas. Our invited audience represents a broad range of stakeholders including international, industry, university, national laboratory and vendor expertise. Our goal for each session is to identify where consensus exists and what questions remain in each of these areas. We **encourage** dialogue during these sessions. Each session chair will at the start of the session state the objective of the session and pose questions to stimulate thinking about the presentations. After the presentations, RES Divisions Directors, acting as rapporteurs, will facilitate a discussion of the questions and how well the session met its objectives. At the end of the meeting, a rapporteur's panel led by Charlie Ader, Director of the PMPDAS RES staff will lead a discussion summarizing the results of each session including the value of current research and what are the next steps. This will be one of the most important sessions of the meeting because it will link the results of the technical sessions and will relate these results to Research goals and strategies. This is your opportunity to influence the future direction of the program—don't miss it.

I want to emphasize that this is not a meeting to discuss specific licensing or regulatory actions but rather to debate the rigor and robustness of the technical basis which underlies these decisions. These discussions transcend organizational and national boundaries and are fundamental to the issue of safety.

There are three ways we are asking you to provide important feedback to RES: (1) The Meeting Evaluation Form which was given to you at registration and which you should complete towards the end of your stay at the meeting; (2) Technical Session Feedback Forms which will be distributed to you as you enter each of the technical sessions. Please complete this form at the end of the session and it will then be collected from you as you exit the room; and (3) interviews with a cross section of attendees who were pre-selected to ensure that all stakeholder groups at the WRSM were represented. However, if you were not pre-selected but would like to volunteer to be interviewed, please see the evaluators in the Rockville Room.

Your input through these evaluation methods will be collected and analyzed by contractor personnel who are conducting the evaluation for RES. The evaluation responses are confidential and will not be attributed to any individual; the results will be reported to RES in the aggregate. Your professional viewpoint and expertise will make an important contribution to the WRSM and to RES activities. We thank you in advance for your participation in the evaluation.

Finally, the overall value of this meeting will be greatly enhanced by your participation in the discussions–we look forward to hearing your views!!!

"Nuclear Regulatory Research in the New Millennium"

by

The Honorable Greta Joy Dicus
Chairman
U.S. Nuclear Regulatory Commission

INTRODUCTION

Good morning, ladies and gentleman. I am pleased to have the opportunity to speak to you at this conference. After looking at the technical sessions on the agenda yesterday, and the talent assembled in this audience, I am particularly pleased to address you knowing that you must have already resolved all of the issues associated with risk-informed regulation and the integrity of primary coolant pressure boundaries! Seriously, I wanted to share with you my thoughts on where our research program is today, and the role of research in the new millennium.

CHANGING REGULATORY ENVIRONMENT

Before I begin my comments about the research program, I need to first describe the NRC's regulatory environment to provide the context for my views of research in the future. As you may know, in the last year, the NRC has been transforming itself, with sweeping changes to many of our regulatory functions. Why are we doing this? We are doing it because the industry's environment is changing, and we must change with it if we are to carry out our mission effectively. We have taken a hard look -- helped by input from our stakeholders -- at the way we are doing business, and we have embarked on a path to change and to improve our regulatory programs. We are seeking greater efficiencies and effectiveness in our processes, and trying to eliminate unnecessary regulatory burdens where they may exist. At the same time, we are continuing to maintain safety and public confidence. This is no small undertaking, and I can tell you that the NRC staff and the Commission have devoted a great deal of time and energy to accomplish it.

We are doing this at a time when our resources are constrained by several years of "rightsizing." I believe that efforts to maintain a balanced Federal budget will continue, which will necessitate that we continue our streamlining efforts. For example, the NRC's FY2000 budget was just approved for the same dollar amount as in FY1999, which means that it actually represents a budget reduction after inflation is considered. In addition, last month I submitted the NRC's FY2001 budget to the Office of Management and Budget, and it is the lowest budget when adjusted for inflation that the NRC has submitted in more than 20 years.

The U.S. nuclear industry has accumulated a great deal of operating experience. The issues that we are dealing with today are more likely to be variations on issues that we are familiar with, rather than the new licensing issues that were present when we were forming our regulatory framework. For the near future, the issues of concern are those associated with aging, renewal of expiring licenses, and decommissioning. Although we have certified several advanced reactor designs, and stand ready to license new power reactor facilities, no orders are projected in the foreseeable future.

As a result of industry restructuring, several difficult issues have emerged. For example, cost-cutting measures and reduced staffing must be done in a manner that maintains safety; the availability of funds for decommissioning must be ensured when companies consolidate or split; the extent of foreign ownership must be considered on purchases to ensure the nation's security is protected; the extent of control by non-owner or contract operators of nuclear power plants must be evaluated to determine compliance with licensing requirement. Moreover, increased numbers of independent system operators supplying power to the North American grid can affect the reliability of offsite power supplies and increase the importance of emergency diesel generators.

NRC INITIATIVES IN RESPONSE TO THE ENVIRONMENT

I turn now to a discussion of some of the more significant initiatives that the NRC has undertaken in response to the changing environment. In each of these, the NRC's research program has been an important contributor to our technical basis and analytical methods.

First, we have just launched a pilot version of our new power reactor oversight program. The new program offers sweeping changes to our inspection, assessment, and enforcement processes. We received feedback from our stakeholders that our processes were too subjective, difficult to understand, and therefore not predictable. In addition, our processes did not adequately recognize the improving performance of the nuclear industry as a whole. The new framework is designed to address these issues. We have worked closely with industry and our stakeholders to develop a concept of "cornerstones"-- key areas of licensee performance that must be monitored to ensure that unacceptable public risks do not arise from nuclear reactor operations. We utilized the results of our ongoing research in measures of performance to develop quantitative performance indicators in each of these cornerstones. This will allow both licensees and the NRC to more easily identify areas that need attention, and to focus our resources accordingly. These indicators, as well as the NRC's current assessment of licensee performance, will be communicated more clearly to the public by posting them in graphical form on our web site (www.nrc.gov) on a quarterly basis. We began testing this pilot program at nine sites in June of this year, and we are optimistic that the program will be able to be implemented for the entire industry in April 2000. Early feedback from licensees on the pilot program is encouraging, but we have more work to do before the program is ready for full implementation.

Another focus area for the NRC has been the renewal of licenses for our older plants, and I am very pleased to report to you on the progress that we have made. We have aggressively worked through literally hundreds of technical issues on the first two applications by Calvert Cliffs and Oconee, and the projected time to review a license has been reduced from over five years to about 24 months. I need to credit the NRC staff for this success story. The staff developed a technical basis for the reviews through research on aging issues, then reached regulatory resolution on the issues by working closely with industry. It really is a good example of firm, fair regulation, while considering stakeholder concerns. In fact, *Inside NRC* published a story recently discussing how licensees are jockeying to be next in line for staff review. So what we have apparently done to reward ourselves is to bring on more work! But I think this a good problem. From a *resource perspective*, the NRC is gearing up to handle this increased number of applications. From a *process perspective*, we will continue our efforts to streamline the license renewal process, develop clear review schedules and milestones, and refine the scope of our reviews. From a *technical perspective*, the NRC staff is examining whether some issues can be resolved generically for all future license renewal applicants, and is consolidating lessons learned from the pilot reviews into revised regulatory guidance that will be published in the next few months.

You may have heard a good deal about "risk-informing" our regulations, but you may not be too sure what that means. In general terms, it represents a philosophy whereby risk insights are considered, along with other factors, to establish requirements that better focus attention on issues commensurate with their importance to public health and safety. Looking back, our regulatory framework was established years ago using experience, testing programs, engineering margins, and a philosophy of defense-in-depth, but without the benefit of quantitative estimates of risk. That framework has served our nation quite well for many years, and we don't expect to throw it out and start over. Rather, we are researching the technical basis for our current regulations, with an objective of reducing unnecessary conservation where appropriate and possibly identifying areas with insufficient conservatism. Specific areas that we are looking at include parts of the ASME Code, In-Service Inspections, improved allowed outage times for technical specifications, and a more systematic approach to fire protection. Is this easy? Absolutely not! But that doesn't mean we should not do it. I expect that we will approach this very carefully, and as our methods of analyzing risk improves, we will continue to refine our approach. The U.S. has taken a leadership role in this area, and I can tell you that the rest of the world is watching to see what we will come up with.

Decommissioning appears to be a growth area. We all recognize that our nuclear facilities are aging. Those that cannot demonstrate their value or are not economical will be shut down and decommissioned. We have recognized that there may be inefficiencies in our current regulatory framework, since we hold our decommissioned facilities bound by regulations that were designed primarily for operating facilities. As a result, in the power reactor area, the NRC is taking a formal look at our whole approach to decommissioning to see if we need to create a new regulatory framework, and to see if we can focus on the areas of greatest risk. Research is contributing by examining various analytical tools and studying the viability of possible approaches to decommissioning, such as entombment.

In developing these initiatives, the Commission has actively worked with our stakeholders to implement new processes that are commensurate with increased regulatory insights, improved industry performance, and continuing advancements in risk assessment methodology. I believe that we have demonstrated the willingness to re-examine our existing programs in a fundamental manner. However, this does not mean we are bowing to industry complaints and political pressures! In all of our efforts, we have not lost sight of our focus on the most safety significant aspects of facilities. We will not promise that our efforts will satisfy all of our stakeholders. However, we are committed to considering all inputs in making our regulatory decisions, and we strive to ensure that our stakeholders understand how we arrived at our decisions. My experience is that even if our stakeholders don't always agree with our decisions, if the process is understood, then their confidence in the NRC is enhanced. At the end of the day, we believe that what we are doing will both ensure safety and provide stability, clarity, and predictability to our regulatory processes. The key to ensuring this happens is having a solid technical basis for our decisions, a basis that is established by our research program.

NRC RESEARCH YESTERDAY AND TODAY

How should research continue to support our initiatives? To address this question, I will provide some historical perspective on our research program. The NRC has funded research on nuclear issues for all of its existence, but not always at the same level. In the early 1980's, the NRC's budget for the Office of Research peaked at over $200 million. At the time, this research supported the development of the technical basis for many broad areas, including Three Mile Island items, severe accident phenomena, formulation of the NRC's Safety Goal and Severe Accident Policies, and modeling of thermal-hydraulic behavior. Many of these endeavors required the use of large scale experimental facilities. Subsequently, the focus of research shifted to issues such as the development and application of risk methods, revising the source term, aging research, and support of advanced reactor design reviews and certifications. However, this research has been less resource-intensive, and with no new plants being ordered in this country over the last two decades, the funding for research has gradually declined.

Today, as I look at where we are, I see that our research program still spans a wide variety of relevant technical issues. We categorize our research into two broad areas. The first is what we call Confirmatory Research, and it constitutes perhaps 80% of our budget. This area supports user needs requests from our front-line regulatory offices, and therefore focuses on current safety issues. This purpose of this type of research can generally be described as to remove unnecessary conservatism in our regulations and to provide assurance that our regulatory judgements are valid. Examples of this in the reactor area includes risk-informing our regulations in 10 CFR Part 50, independently reviewing industry operating experience (a function previously performed by the old NRC

Office of Analysis and Evaluation of Operational Data, or AEOD), ongoing research into structural and geological engineering issues, and radionuclide transport and health effects.

A second area of NRC research is called Anticipatory Research, and it constitutes the remaining 20% of our research budget. The purpose of this type of research is to anticipate future needs, and to provide the technical basis to support future regulatory actions for emerging safety issues. Examples of this type of research include addressing PRA limitations as the NRC transitions to a risk-informed regulatory process, development of risk-based performance indicators, assessing links between performance and plant safety, and deregulation and its impact on plant safety.

From a program perspective, I believe that we are focusing our research in appropriate areas, and we are anticipating our future needs. From a resource perspective, we are operating with a FY2000 budget for research of around $40 million. We are actively pursuing opportunities to leverage our research funds through cooperative efforts. We are prioritizing our research activities in consideration of risk, uncertainties, and future challenges. And yet, I feel that we can do more, and I will elaborate on that in just a minute.

RESEARCH IN THE NEW MILLENNIUM

What is a vision for research for the new millennium? The challenge in answering this question is to be able to successfully project yourself into the future based on trends today. Of course, if I could do that consistently, my stock portfolio would be much healthier than it is, so you must treat any predictions with that fact in mind. Nonetheless, I shall attempt this rather lofty goal.

For trends, I think the industry is maturing and will focus on optimizing their current plant configurations rather than developing new and innovative designs. I also think that industry consolidation will continue, thereby reducing the number of utilities as well as the number of companies supporting the utilities. In addition, commercially available parts and hardware may be used more often rather than parts with a long Quality Assurance pedigree. Finally, the use of computers for modeling in lieu of actual experimentation will likely increase.

The NRC has already taken action to address some of the trends, and these are the new NRC initiatives that I had previously described to you. But these are just the start. New technology, such as advanced instrumentation and controls, can certainly have an impact on plant safety. For example, advancement in computers and information technology are coming at a rapid pace today, but research is needed on the reliability of this technology before it can be widely applied to nuclear power plants. Advancements in fuel design and materials are an emerging area, particularly the use of high burnup and mixed oxide fuels. In addition, although the NRC is nearing a decision on issuance of the first renewals of licenses, research into aging and associated materials research will continue. Finally, risk-informing our regulations will require research to establish a sound basis in both technical issues and probabilistic risk assessment (PRA) techniques. I must also briefly mention high-level nuclear waste disposal, which remains a difficult problem that will only be resolved with continued research. Let me say that the Commission remains firmly convinced that a permanent geologic repository is the appropriate mechanism for the U.S. to ultimately manage spent fuel and other high-level radioactive waste. We are continuing to develop a Yucca Mountain review plan and to resolve key technical issues to prepare for reviewing the DOE license application expected in 2002.

Earlier I said that I would elaborate on ways I thought we could continue to improve our research processes. I believe that we must reassess the way we do our research, just like we have done in other regulatory areas. Let me say at the outset that I believe in the value of research, and believe that the budget for it should be maintained as a minimum, and perhaps should even be increased. As a regulatory agency, we must preserve our independence and maintain a broad perspective to fulfill our mission of maintaining safety. Nonetheless, I also recognize that the environment is changing, and we do not have the ability to conduct extensive exploratory research. Long term research has a place, but many things today do not lend themselves to that. Instead, we must develop feedback mechanisms so that our programs can be continuously examined to ensure that the research is relevant. We must develop and refine our prioritization processes to ensure that our resources are being focused on the most significant issues. We must ensure that our research is linked to the needs of our stakeholders. In other words, our research programs must have a certain agility to respond to the environment.

Our research programs must be timely and responsive to both internal and external stakeholders. Too many times I have seen a well-thought out and well-executed research project completed, but not really used because it was either not timely or not responsive to user needs, or both. I recognize that high quality research takes time, so the challenge is to focus our available resources in a way that ensures a quality product in a timely manner. In addition, we must emphasize delivering products that contain recommendations for applicability. Again, I cannot tell you how many fine two-inch thick research projects I have seen that do not provide relevant recommendations and leave it up to the reader to figure out how the research should be applied. One way to improve our programs is to adopt the approach the NRC has learned in responding to the changing environment: listening carefully to its stakeholders. We recognize that our stakeholders have very valuable insights, and we have also found that they are not bashful about volunteering them! These insights can be used to help focus our resources and to shape our efforts in the future.

My vision of the Office of Research in the new millennium would be a center of excellence and source of expertise. This center would maintain a cadre of reactor safety specialists in various key areas, with independent and unbiased expertise across a broad spectrum of advanced nuclear technology, to provide the technical basis for robust and transparent regulatory decisions. Experimental facilities and resources would be maintained to ensure our ability to respond in a timely manner to new or emerging issues. The Office would complement the front-line regulatory activities of the agency and independently examine evolving technology and anticipated issues.

Finally, new and creative approaches to research will increasingly be used. Partnerships with industry, foreign organizations, and other government agencies will become more common. Our joint research with the European Union, and the recent Memorandum of Understanding with DOE on Cooperative Nuclear Safety Research are good examples of this. As the costs of large-scale experimentation rise, we will have an increased need to leverage the work of others, even while maintaining our necessary independence on regulatory matters.

CONCLUSION

I would like to close by noting that Dr. John Ahearne, a past Chairman of the NRC, recently headed up a comprehensive review of NRC processes for the Center for Strategic and International Studies. The review was co-chaired by Senators Pete Dominici and Bob Graham, and Representatives Joe Knollenberg and John Spratt, and received inputs from a wide range of organizations in the nuclear field. During a briefing to the Commission on the final report, Dr. Ahearne noted that research must be a vital part of the NRC's programs. I consider that to be a good example of valuable and timely input from our stakeholders, and I would like to emphasize that the NRC needs continuing input such as this to help us shape the future of our research programs. I believe that this input is essential to furthering a mutual understanding of the issues affecting the safety of nuclear power. I appreciate the opportunity to speak with you today.

Now I will be glad to answer any questions you may have.

How Best to Focus Both on Safety and Burden Reduction -- The Research Role

Margaret V. Federline, Deputy Director
Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission

Good afternoon. Welcome to our second expert panel on how best to focus on safety and burden reduction, the research role. For me this is a really exciting part of the meeting, because it gives us an opportunity to bring to bear the perspectives that we've heard from all the technical sessions. We will able to reflect on these and understand what they mean for the themes and objectives of our meeting overall.

We've all benefitted from the sessions that we've participated in, and there have been a wide range of papers and often some very lively discussions, which I think is healthy in a meeting like this. We've heard from the NRC staff on how a risk-informed approach can optimize the balance between safety and burden reduction. We've heard about the potential for late-blooming issues which could impact safety through vessel embrittlement. We've heard from the NRC staff on improving processes, like streamlining the generic issue process, which could reduce unnecessary burden while not impacting safety, and we also heard a very energetic discussion by the fuels folks about what's an appropriate approach to ensure safety in the use of high burn-up fuel. So I hope some of these papers will be in the back of your mind as we proceed with our discussion today.

We're very fortunate to have with us this afternoon five recognized experts who'll bring diverse perspectives based on their extensive experience in utility management, regulation, and research. I just wanted to set the context for our discussion.

As you heard from Commissioner Merrifield and other presentations, NRC has embarked on a top-down planning process to assist us in aligning our goals across the agency. We believe that this process will help us in setting our priorities and measuring our success against these goals.

Today, our topic focuses on how to balance maintaining safety and reducing unnecessary burden. I think the first thing that we have to ask ourselves is what do we really mean by maintaining safety? I think if we probably went around the room, there would be a number of different interpretations about what we mean, but we at the NRC have really seen an improving trend in U.S. utility performance over the years, and we want to make sure that that trend continues, not just in the near future but in the long future, as well.

Now, also included in our definition of maintaining safety is the face that, if significant increases in safety are identified and can be justified under our back-fit rule, these will be incorporated under maintaining safety, as well. Now, there are different factors which are influencing the goals that I've just discussed.

Economic deregulation of the industry is driving higher productivity from existing plants, and this also includes pressure to reduce the cost of regulation. To achieve higher productivity, utilities are looking to extend fuel burn-ups, to upgrade power at some plants, to expand on-line maintenance, and to improve shut-down activities, as well. Now, as you've heard at the meeting already, in order to make our regulatory processes more effective and to enhance our focus on safety, we're moving to a risk-informed approach, and I do want to emphasize that this is risk-informed and not risk-based. Also, we need to think about the maturing of the industry and what that means in terms of the need to understand our existing margins and how that could affect safety for the future.

Now, to be an effective regulator, we have an obligation to ensure that margins are adequate for safety, but we still have an obligation to ensure that we are not unnecessarily impacting the industry's competitiveness in the marketplace. We must understand margins well enough to ensure that they encompass the broad range of uncertainties that are inherent in the systems that we regulate. So, what approach might we consider in reducing unnecessary burden while maintaining safety? I have identified several elements of an approach. I hope we'll have some discussion on these as the panel makes its remarks.

One element might be to move away from bounding analysis with very conservative margins, what we've typically known in the past as deterministic regulation. We may want to perform realistic analysis specifically considering margins in the acceptance criteria. If this is the case, though, we need to make sure that each important assumption is evaluated and sensitivities are understood before we take this course of action. As we make burden reduction decisions, we should also be evaluating the cumulative effects of small designer operational changes. I know there are various viewpoints on whether that is a good practice. It's also important as we move to risk-inform regulation that we have a robust basis in our risk-assessment methodologies. We must improve PRA technology and standardize those methods, as well. And another approach that we might take is to clearly define what we mean in terms of defense in depth as it relates to a risk-informed approach. I personally believe this is going to essentially give us the acceptance criteria that we need in moving towards risk-informed.

Now, each element of this approach drives the need for robust technical basis, including research. I'd like to pose just a few questions to stimulate our discussion, our panel, and I hope we will have an active dialogue with the audience, as well. What is the role of research in this new paradigm? Have we been premature in some cases of doing away with research in areas of low probability? What enhancements to PRA technology are warranted to ensure that goals are met?

Although strides have been made in risk-informed regulation, we still need to be cautious and recognize that there are some limitations in the technology. For example, the treatment of external events and human error, among others. We also need to ask ourselves what technical and analytical uncertainties must be pursued to ensure a sound and realistic technical basis. We heard some of that discussed in the sessions on pressure vessels and primary boundary.

Will a more robust technical basis maintain public confidence even though we are moving towards reducing unnecessary burden? I think stakeholder reaction is a very important consideration as we move towards reducing unnecessary burden. We also have to ask ourselves what minimum research capabilities need to be maintained, to answer the inevitable safety questions that history tells us will arise. What initiatives to improve operating efficiency or use new technology require research to provide a basis for regulatory acceptance? We heard a number of these discussed in the session on new technology. There are a few additional questions that need to be considered which are associated with our research activities.

What does operating experience reveal about the coincidence of maintaining safe operations and reducing unnecessary burden? Does successful performance mean that we can reduce the rigor of our oversight process, or does it just tell us that our oversight process has been effective in producing the needed results? What can be learned from experience in other industries about both maintaining safety and reducing unnecessary burden? How does one assess the risk trade-offs between maintaining safe operations and reducing burden? And will new technologies reduce unnecessary burden or margin?

I heard an interesting paper in the session on "New Technologies" on safety assessment techniques for digital safety systems, and it seemed to me that this work would both improve safety and reduce unnecessary burden.

With those thoughts as a backdrop, I would like to turn to our expert panel now, starting with Dave Helwig.

How Best to Focus Both on Maintaining Safety and Reducing Unnecessary Regulatory Burden

Dave Helwig, Senior Vice President
Nuclear Services
Commonwealth Edison

Thank you very much for the introduction. I'll make one correction to it. I am no longer the Interim Senior Vice President of Transmission and Distribution. I'm back in Nuclear. I actually was talking to the EPRI Power Delivery Advisory Council the other week, explaining to them all the things that I'd learned, trying to straighten out and diagnose the problems that we had in our T&D system at ComEd this summer, and at the end of that interesting story, I said "and boy am I glad to be back in Nuclear," and they all laughed at me.

I'm glad to be back in Nuclear, love the technology, love what we're doing, find it a very exciting place to be. I just go do this T&D thing once in a while as a hobby when utilities I'm with get in trouble, I guess. So, anyhow, it's a pleasure to be here. I really was very pleased when Ashok asked me to come and share my thoughts and perspective on this subject.

I've attended this Reactor Safety Information Meeting in the past, years gone by, with a bit of a technical slant or perspective being brought to it based on my experience and role. It's traditionally been research-focused, something I've been interested in, but yet I'm not a researcher, haven't been a researcher.

I do love the science and believe it's very foundational and important for us, but you know what? Today, it is for our industry all about how research fits the business model that we're in, and that's very much what we are in. We've been moving towards it gradually. We're now moving towards it at a very accelerated pace as a result of the deregulation of the industry that's going on. I think the remarks, the perspective that you had up there was right on the mark: economic deregulation, how are we going to risk-inform, and the aging of our plants.

As a businessman responsible for a large fleet of nuclear plants and a fleet of nuclear plants that's going to become even larger as we continue to merge and consolidate, that is what it's about. We would not be pursuing a merger with another utility that had a lot of nuclear plants if we didn't think it was all economically viable. That's first and foremost a consideration today.

We would have sold, gotten rid of, or closed down Commonwealth Edison's nuclear plants if we didn't believe it was viable for us to make a substantial profit in running them well. So, for those of you that don't know -- maybe you've been somewhere else or not paying a whole lot of attention -- Commonwealth Edison has currently operating 10 nuclear units, 10 large nuclear units, the largest fleet in the country, certainly, and amongst our distinctions, besides over the years having the largest fleet, has been having probably the poorest-run fleet of plants. Now, that was -- to those of us that have come to Commonwealth Edison to turn around performance -- that's what we started with, and we don't have the same perspective on justifications and rationalizations of why that was the case.

We have, in fact, been able to improve performance substantially, but the truth of the matter is, the reality is that, if we had not been able to improve performance and operate those plants reliably, they would have been shut down, because we could not -- as quickly as we could bring in additional capacity, either purchased or installed, to replace them, because you couldn't afford to be in the business without them being run well.
So, I'm pleased to say we're running them quite well, and in fact, since we already have the disease and we know how to run them well, we are in the process of merging with other utilities that have substantial numbers of nuclear units and aspire collectively to collect even more of them under our operating umbrella.

So, what's it all about, being in this business? It's operating the plants reliably and cost-effectively. The very next line item you come to on the list is managing the assets. Maximizing the value of your assets is the business model that we really have to be attentive to, and that brings to bear a number of things. How do you maximize the value of one of these nuclear plants? .Well, in fact, you have short outages, as short as you can have them, which means you do the maintenance on-line, as much as you can figure out how to do it. You have to do it right. You've got to do it right, but that's the way you get the most generation out of it.

Gee, what's the next thing I could think of? How about increasing the power output of the units? Turns out, for 5, maybe up to 10 percent, typically the balance of plant has enough margin in it to handle it. So, we've had many, many utilities pursuing power upgrade, again just to generate more power from the assets we already have. The name is cents per kilowatt hour, and we all know that it's a whole lot easier and there's a whole lot more leverage working on the denominator of that simple equation than there is working on the numerator. We can drive these plants into cost-effectiveness by increasing their generation much more easily than we can reduce our expenditures on them.

The next thing it would lead you to is life extension. You want the plants not only to operate well while you're operating, but you want to be able to run these plants longer, again just to get more out of the asset that we already have. So, when we look at what is the role for research to play, for science to play in this business model, I think it brings a couple of interesting insights.

I don't pretend to have all the answers. I have a couple of thoughts, some that I feel very strongly about. I will start with administrative burden, a "gimme". Now, at first, that seems to be pretty simple, but I submit to you that there are a number of administrative requirements that we have, reports that we're required to make, records we're required to keep, that are based on a premise that could be better understood and eliminated, maybe even something as simple as the record keeping for 10 CFR 50.46, which is anytime we have more than a 10-degree change in our predicted peak clad temperature due to some plant change or change in analysis or type of fuel, we've got to report it.

There's probably -- for every two-unit site, there's probably some engineer who's working almost full-time keeping track of all that and worrying about that. I have a feeling that the tolerance band can be broadened. So, it's not all that it appears to be.
I submit to you that, even in the area of administrative burden, there are benefits to be gained by going back and looking at our basic science, our basic understanding of the regulations.

A second issue I would point to is emerging issues. Well, I sure can't extend the life and continue to operate these plants as long as I physically can if I don't understand what issues there are that might come up, whether it's reactor vessel embrittlement or some other kind of materials issue or any of the plethora of issues that we're looking at as we examine life extension. It is very important to us in our business model to understand those aging mechanisms, to be able to practically manage them to extend the life of our plants and, therefore, maximize the value of our assets. This is a very high priority for us.

We have five business strategies in our business planning model at ComEd. We did at PECO, and most of the other companies do as well. The ones that are really focused on the long haul have as one of those four or five top business strategies asset management, all these things about maximizing the value of those assets, including extending life.
Fuels -- let's talk about fuels. Well, for a well-run and operating plant, typically the cost of fuel will be about one-third of your cost of generation. Fully a third of our cost of our generation is in fuel cycle cost. So, how do you leverage that?

Well, higher burn-up cores, higher density cores, higher power levels all enable you, again, to maximize the economic equation that we have to be focused on, always focused on. So, the fundamental science of understanding fuel behavior, understanding how to get more out of the fuel, how to get more robust designs that will operate more reliably are very important in this economic model.

And lastly, I'll turn to my favorite subject, which is risk-informing what we do.
As was mentioned, I chair the NEI Risk-Informed Regulation Working Group, and it's a pretty big challenge as a matter of fact, because of the business model that we're in.
I've been involved for quite a few years in the risk-analysis arena, going back to being a data collector for WASH-1400, as a matter of fact, which was done in our Peach Bottom unit when I was at PECO, and remain very involved today. But today I find myself and a few others, as senior managers in this industry, having to work very, very hard to maintain the interest level of our peers at other utilities that are responsible for nuclear operations on this whole risk-informing, PRA, risk analysis world.

We're having to work very hard to maintain their interest, because the economic benefit to come out of this endeavor is not entirely obvious. There are no guarantees. We're all trying to figure this out. I believe that there are many in the industry, including myself, who would love to see this work out to our benefit, to have it make a substantial difference in the way we manage, the way we regulate, and the bottom line, the cost of our operation, but there is not, at the moment, a clear path to success.

We have some models that we're working on, we have some pilots that we're trying to get organized, and the truth of the matter is I have a very hard time generating enough interest in utilities to put up even the seed money to be a pilot because of the stress on their resources. The business model requires them to limit the resources and only fund and invest in those things that will have maximum payoff.

So, our challenge in the risk-informed world is, in fact, to demonstrate that the risk-informed regulation thrust will pay off in the business model that the utilities are in today.

So, what does that require technically? I mean that's a nice philosophy to state, very important one to keep in mind, but what does it mean? It means, in my mind, when you get right down to it, what we come up with as a framework to go about doing this needs to be practical. It needs to be able to be applied. It needs to be focused on specific benefits to be derived, not just the science but the science in pursuit of those issues. In fact, as I reflect on my experience in risk analysis over the years, I find that most of the problems that we've had have been lack of rigor in the analysis, lack of thoroughness to consider the plant's inherent capabilities than the actual science or phenomenology itself. So, maybe that's just a different kind of science. It's how you apply the powerful analytical tools and techniques that we have to find out what it is that is important. How do we get a fix on what the actual level of risk that these plants represent is? What are the technical issues that dominate that risk? Because we want the wiggle room, we want the flexibility to be able to make trade-offs.

Currently, if I look at the risk profiles of our fleet, don't hold me to the accuracy of this, but I would estimate that about one-third of the plants as currently assessed would have an identified or perceived risk that is higher than the threshold, a higher risk than the threshold at which our current thinking about allow to pursue the benefits of risk-informed regulation.

As I look at the distribution of those, it so happens that there are a few technical and phenomenological issues that dominate that profile of risk. So, I think the challenge for us is to develop the framework to do the analysis and the technical understanding to really enable us to accurately characterize what the risks are.

That's my complete thoughts on the subject at the moment. I'm very, very excited about the potential that is before us, and I truly hope we're able to fulfill the promise. Thank you.

27th Water Reactor Safety Information Meeting

Bethesda
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MAINTAINING NUCLEAR SAFETY AND REDUCING UNNECESSARY REGULATORY BURDEN

JUKKA LAAKSONEN
DIRECTOR GENERAL
STUK-FINLAND

Maintaining nuclear safety

What do we mean by that?

Can the current safety level be considered acceptable to day, and also in the foreseeable future?

If yes, is it realistic to aim for a steady-state safety level (stagnation)?

Reducing unnecessary regulatory burden

Can we identify the dominant unnecessary burden?

Burden relevant to research:

- excessive safety margins?
  - but what about the safety margins the owners would prefer to protect their investment?
- irrelevant regulatory requirements?
- too much emphasis on issues that are less important for safety?

Burden not very relevant to research:

- inefficient regulatory processes that delay or hamper licensee activities?
  - excessive bureaucracy
  - poor communications between licensee and regulatory staff, and also inside the regulatory body
  - lack of service-minded attitude among the regulators
- business habits within the industry not well adapted to regulatory processes?

Acceptability of the current safety level

Safety level, as it is believed to be, has been found acceptable during the licensing process.

- In a situation with no increase in the number of the plants, one could argue that there is no need to draw other conclusions.

However, uncertainties exist and new concerns are occasionally emerging from the research and operating experience.

Research is needed to explore uncertainties and issues that are still unknown. Recent examples are:

- High burn-up fuel response to transients
- Boron dilution scenarios
- ECCS sump / strainer performance

A steady-state safety level is not a realistic goal

Key element in maintaining safety is maintaining full understanding of each relevant safety issue, by the nuclear community at large and especially by the operating and regulatory organisations.

Retaining the tools and documents developed during the earlier "hey-days" is not a feasible approach, and the risk of misapplication by incompetent people increases with time.

Competence of the entire nuclear community will inevitably degrade unless the most talented people ("leaders") can be kept in the field, and a new generation of "leaders" can be attracted.

- "Leaders" will be around only if new challenging problems are presented to them to be solved.

New challenges, with high ambition levels, could be related to improvements in nuclear power plant

- safety,
- efficiency, and
- economics.

A common European approach is a constant drive for improved safety level at operating plants

This approach is considered necessary because of its importance for maintaining competence and wide understanding of key safety factors.

Safety improvements often go hand in hand with improvements that lead to improved economics:

- Reduction in the frequency of anticipated transients – this reduces core damage frequency and improves overall plant performance.
- Power upgrading – this usually requires improved control and protection systems, and new additional safety features.

When a new concern arises, it is sometimes cheaper to make evident safety upgrades than to prove by research means that changes in the original design are not needed.

Attempts to reduce regulatory burden by reducing safety margins may be dangerous

Key fields relevant to nuclear safety have reached maturity – in terms of ambition levels set three decades ago.

World-wide competence in the reactor physics and thermal hydraulics has already passed its peak level, and accelerating decline is imminent unless new ambition levels are set.

Here is a contradiction: a reduction of the excessive safety margins, and a move from bounding analysis to more detailed ones requires improved understanding of separate physical phenomena, their coupling, and improved tools.

- Development of improved tools (such as 3D codes with coupled reactor dynamics and thermal hydraulics) provides a good challenge, and thus supports maintenance of competence.

Knowledge on the original design basis criteria and their basis is of crucial importance.

- For instance, the nuclear fuel used to day is quite different from the fuel that was used in the early 70's, when the safety criteria and safety margins for fuel were developed.
- Gradual design changes should be supported by research, either to prove the continued validity of the original criteria or to develop new more appropriate criteria.

Role of PRA in reducing regulatory burden

PRA provides insights and a sound support for educated engineering judgement.

- Setting priorities
- Pointing out items in need for improvement
- Optimisation of Tech. Specs. and on-line maintenance (from risk perspective)

PRA should not be used for explaining away safety concerns from rare events.

- Prediction of passive component failures by PRA means is not accurate and reliable.
- Need for safety features that provide defense-in-depth against rare events (large break LOCA, core meltdown) can not be decided from PRA results.
- Containment performance at TMI-2 was an example of a safety feature that provided a good protection in an unpredicted event.

PRA results (CDF) are not testable, so their validity is difficult to ascertain.

- Conservative deterministic safety analysis tends to stack safety margins.
- PRA tends to stack uncertainties.

How Best to Focus Both on Maintaining Safety and Reducing Unnecessary Regulatory Burden

Tom Murley, Consultant

It's good to be back here and see so many old friends, and thank you, Ashok, for the invitation and the opportunity to do that. It is hard for me to keep up with all the changes that are taking place at NRC these days. I'm going to make a few comments about some of the changes that I see, and then, at the end, I'll focus on the topic at hand, which is research and the importance of research.

I should add to the introduction that I did direct the reactor safety research program in NRC from 1975 to 1980, and I see Bob Wright down there who was with me in those days, and probably some others.

The new reactor oversight process, for example, one of the big changes, as I understand it, involves dropping the SALP, dropping the watch list, changing the enforcement policy, which was greatly needed, in my judgement, and reducing the inspections at most plants and moving toward an objective performance indicator-driven oversight process.

I do support this, by the way. I think it's going to be a challenge for the staff, because these objective performance indicators are generally viewed as lagging indicators, and that means that, typically, there can be a substantial decline in the indicators that won't show up until a year or two after the underlying organizational problems or safety culture problems that caused them are evident.

So, the staff, I think, is going to have to adapt, let's say, to use this new system, but again, I'm confident in the staff. I think they have and always have had common sense, and there, of course, will continue to be resident inspectors who can spot these negative trends.

This was a change in the oversight process that was needed. There's no question. The old one had outlived its usefulness, and one had to recognize the improved safety performance of the plants in the U.S. Not only has the average safety performance improved over the last decade, but the spread has gotten a lot narrower, so that the grouping of performance around the mean is much narrower, I think, and in that sense, in my judgement, this new oversight process can be made to work.

Secondly, I'm particularly pleased to see the move toward risk-informed regulation. This I view as the best means to focus both the NRC staff attention and utility operators' activities on those issues that are most important to real safety. I am not terribly familiar with the proposed new rule, 50.69, although I've read about it, but it seems to me to be an innovative approach for risk-informing Part 50. I must confess I did not think of that approach, but I think it can work. It's, as I say, very innovative.

But there's going to be years of work ahead. It's not something that can be promulgated next year, if I understand the approach to the Appendix T correctly. Appendix T is how you decide what systems, structures, and components are important for an individual plant. So, with that said, I think there are some cautions that I would like to throw out as we move down this path in this country to performance-based oversight and risk-informed regulation.

The first caution is that risk-informed operation has to go hand in hand with risk-informed regulation, and I am dismayed that it does not seem to be getting as much attention, either by the industry or by NRC. Most of the discussion seems to be focused on the relaxation of regulatory requirements, or burden reduction, as it's noted here, and I think that's regrettable.

If risk-informed regulation comes to be seen by the public as simply a code word for deregulation, then I think it's ultimately doomed, because it won't have public support in the long run, and it doesn't have to be this way. I fully support Jukka Laaksonen's questions and his answers, too.

What do we mean by maintaining safety, and can a steady-state level of safety exist? I agree with him. I don't think it's practical that you can talk about a steady-state safety level. But if "maintain safety" means something along the following lines -- that, it isn't necessary for the regulator in this country, NRC, to take any major new initiatives to improve safety, then I think that makes sense.

On the other hand, with risk-informed operation, if the industry can adopt it and move in that direction, then one can have, with risk-informed regulation and risk-informed operation, two sides of the same coin -- I think you can have improved safety and reduced burden at the same time.

There is, in fact, a good deal of risk-informed operation that's already going on in this country to improve safety. Some utilities have years of experience doing this. I think you've probably heard from Steve Rosen from South Texas. He has been saying these things for years.

Other utilities are quite advanced in using PRA to risk-inform their every-day operations, and I must say some utilities are doing practically nothing, and that's kind of the problem with voluntary efforts in this country, always has been, as a matter of fact. Let me give a couple of examples.

Could I have the first slide, Paul?

This is an example of the results of outage risk management. It is essentially a learning curve for a 1,200-megawatt BWR. This happens to be Grand Gulf. They do a very good job of risk-informed operation. And what it shows is, for a series of four refueling outages -- now, this goes back to the early '90s. I think the last refueling outage that they talk about was 1995, on this chart. This is the last information that I have. But they plan their outage -- let me take a minute just to give an outline of what they do.

About a year or so before the outage starts, and certainly six months before, they run an outage risk assessment and management code -- this is an EPRI software package -- that plans the major blocks of work. Then, as they get closer to the outage, they keep running this software package, which reduces the risk during the time of the outage from different work activities.

About several months before the outage, they take a totally different approach. The OSSG, which is an on-site safety group, does a deterministic safety assessment of the schedule, once the major activities of the outage schedule are fixed, and here they focus on the five key safety functions -- inventory control, shutdown cooling, AC power availability, reactivity control, and containment control. So, this is a deterministic look, then, at these safety functions.

And then, third, they look at certain accidents that can happen and ask themselves, well, what can we do to mitigate these? So, they have on hand some contingency plans and some recovery plans for accidents. So, there's -- you could almost say they look at shutdown safety from three different angles, and I think they've got it covered very well.

And then a fourth aspect that they do is, at the end of the outage, they do a retrospective analysis of how well did they do. This is a sign of their success.

Plotted on the vertical scale is the integrated risk that they calculated. It's in terms of events per year, but it's -- I don't know if it's boiling events per year or core damage, but it doesn't make any difference to my point. They integrate that arrive at an integrated risk for that outage. For the four outages plotted, they have reduced their outage risk by a factor of 200 over four outages.

So, this is, I think, a dramatic example of what can be done by risk-informed operation, and it bothers me that this is not being advertised as much as it should be, and it's probably not being used as widely as it should be either.

Second chart -- this is another example of using risk-informed operation. I'll have to explain it, but it's very simple. This is a weekly chart, a plot of the risk, at a 1,100-megawatt PWR in the northeast, and it's for the week of October 17th to October 24th, so it was last week. This is the kind of information that's presented at their morning meeting of the activities that are going on the following week, and you can see there's Monday, Tuesday, Wednesday, whatever.

Over the years, they've gotten so good that there are not major peaks in the risk. It used to be several years ago, I would see values of risk during a certain day that were 13 times the baseline level of risk, and these were all allowed by the tech specs, of course. But they've gotten so good at it over the years that you hardly see any major peaks.

One of the activities that they note down there, recommendations on Wednesday, for example, don't perform solid-state protection system testing while you're doing switch-yard evolutions, because there are some switch-yard breaker calibration checks that were going on at the same time.

Now, you might ask -- these are pretty common-sense things to do. Yes. But unless you've got a structured way, like PRA, to focus in on these activities, you could very well miss them, and furthermore, this is a way to get all the people on-site singing off the same song sheet. I am convinced that this activity, these kinds of risk-informed operation activities, are a major reason why we've seen the overall safety indicators improve over the past ten years.

It seems to me, though, that NRC should be touting this a little more, should be giving at least as much attention to these safety improvements as the burden reduction aspects, although they both go hand in hand. I think you can have them both. Now, there's another caution that I'd like to mention about using risk-informed regulation.

Could I have the next chart, Paul?

I was asked to take part in an IAEA workshop in Karlsruhe, Germany, this past June, and this is a chart that I used. It was primarily for eastern European countries, but we talked about a lot of things, and one of the things they did talk about was risk-informed regulation, and I threw out a caution, and it's much the same as Jukka's cautions.

Let's be honest. You cannot measure total safety risk of a nuclear plant, because you can't predict human performance, and you cannot model the pervasive effects of safety culture at a plant. Now, that's not to say that PRA has no use, obviously. I am a fan of PRA. But use it where it's applicable, in things like risk-informed operation and even aspects of risk-informed regulation, but you can't measure total safety risk. So, you're always going to have to rely, in the end, in my judgement, on maintaining a very strong safety inspection and oversight program at these plants.

Okay. Now, I'll turn to research. I am a big fan of research. I always have been. I think, frankly, if I were king, we would have research just for the sake of research. I think you need it. But I recognize, in these austere times, that we have to probably justify it more carefully. But all of this activity on risk-informed regulation and risk-informed operation would not be possible if somebody hadn't done the research 25 years ago, if there hadn't been WASH-1400 and Saul Levine and Norm Rasmussen who invented this technology.

I do get dismayed by people who think that you can pull a rabbit out of a hat and don't realize that somebody has to put the rabbit in the hat, and sometimes it takes 10 and 20 years to do that, but all I can say is we have to keep trying, and if I can help, I'll be glad to try.

Research also nurtured this technology for 15 years, and now it's bearing fruit, and it has, I think, for the last decade or so. But there are areas, some of them mentioned -- I think Margaret's questions framed the issues very well. There are some that I would add, and I think maybe you've mentioned the standards for PRA.

Clearly, as we moved into risk-informed regulation and operation, we need standards so that you can believe the results that you're getting. One has to be careful, because PRAs don't have to be perfect to be valuable and usable, for that matter. In fact, PRAs will never be perfect, in my judgement, for these very reasons, that you can't model some aspects of the risk in a plant.

The new Appendix T that is to be promulgated to risk-rank structures, systems, and components is going to need a lot more work, I think. One has to be very careful in using the single-event importance measures that we have, the Fussel-Vesely and risk-achievement worth measures. They're probably not appropriate, actually, for redundant, high-reliability systems, because they'll always show low importance, and we know that that can't be right, or we have to look at it a little differently. So, I think, in one area, Appendix T is going to take a lot of work, and I think it's going to take research work to look at better ways to rank structures, systems, and components.

We're moving into an era of economic deregulation which is going to change the way we think about a lot of aspects of operation. I'll give one example. A plant I'm familiar with is now in a totally deregulated economic mode. Every day they bid in a price for the power that they're going to produce the next day. It's usually zero, by the way. They bid in zero so that they can be sure of operating.

But some days, the market clearing price is $1.50 a megawatt hour, like last week in New England. Some days, in July, it's $1,000 a megawatt hour. Now, what do you think when you're operating this plant? Are those hours equal? Of course not. You've got to be very, very assured that your plant's going to be ready to operate in July and August. In fact, they'll probably make most of their profit in three months of the year.

The concept of capacity factor over a year may not make economic sense, because the plant has got to be operable during those three, maybe four months when you're making all your money. So, on-line maintenance is taking on a different aspect, and I think -- my judgement is people are going to rethink their whole approach to maintenance, and maybe we'll have spring and fall shutdowns just to be sure they can be operable in the summer and winter. Higher fuel burn-up, power upgrades, we heard about. Both of these can reduce safety margins.

I think soon you're going to be getting into areas where maybe our regulatory limits are not applicable anymore. Have there been reactivity insertion accident tests that cover these high burn-ups? I don't know. Maybe it's being done in France. But I think, soon, the legacy of NRC's safety research program is going to be out of date, and so, at the very minimum, there has to be some research expertise in this country that, if it doesn't actually do the tests, can at least follow and interpret for our own purposes what's being done overseas.

I really wish you luck, and I will remain a fan of research.

Thank you.

The Role of Research in a Risk-Informed Regulatory Structure
Some Thoughts

D.A. Powers, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission

Today we are discussing at Ashok Thadani's behest the role research will play in reducing regulatory burden. The first thing to note about this topic is that it does include research done by the Nuclear Regulatory Commission. But, it also includes research that will have to be done by the licensees. The second thing to note is that safety in the use of nuclear energy to create electrical power is a burden. It is a burden on both the NRC and the licensees. The licensees have accepted this burden when they accepted the offer by the government to make commercial applications of technology developed in government laboratories. Reduction in burden may not be the best name for the topic of our discussions.

Some will say that it is not burden that will be reduced. It is "unnecessary" burden that we seek to reduce. One has a very hard time identifying "unnecessary" burden. What looks to be one man's burden is another man's prudent measure taken in the name of defense in depth! Defense in depth is often cited as the underlying safety philosophy of the regulations imposed by NRC on nuclear power production. It is assuredly an expensive safety philosophy. It is my belief that the defense in depth philosophy was devised in the early days of nuclear power production because:

- there was very little experience with nuclear power production,
- there were no widely recognized standards for safe operation of nuclear power plants,
- there were large uncertainties in the estimating the consequences of accidents at nuclear power plants, and
- an accident at any nuclear power station would have grave ramifications concerning the continued viability of all commercial nuclear power plants.

To be sure, the base of experience in the operation of nuclear power plants has grown substantially over the three decades. It is not clear, however, that the other reasons for adopting the defense in depth safety strategy have disappeared.

The early development of safety standards for nuclear power plants was based on concepts rather similar to the "unit operations" approach to chemical engineering during the era. Because there was no experience with the whole of a large nuclear power plant, the plant was divided into distinct systems, structures, and components that could be analyzed in detail and safety experience derived from analogous systems in other industries could be applied. Defense in depth was applied then within each of the major systems of nuclear power plant.

We can come here today and question the regulations developed in the past because of capabilities that NRC-sponsored research has given us. NRC research has invented probabilistic risk assessment or to some probabilistic safety assessment. Whether it is called PRA or PSA, NRC research has invented a method that allows so complex an entity as a nuclear power plant to be examined as a whole rather than as a collection of individually optimized subsystems. (It is interesting to note that in chemical engineering optimization of "unit operations" has fallen into disfavor relative to optimization of chemical systems as a whole.) It is not surprising that this superior technical capability makes it possible to question the utility or even advisability of safety measures defined in the past with more limited analysis capabilities.

Indeed, we see the Nuclear Regulatory Commission taking steps to advance the use of this new technology. Certainly they are making it inviting to apply PRA methods to support requests for changes in the licensing bases of nuclear power plants. More dramatic changes in the whole regulatory structure for nuclear power plants are being considered. Companies that specialize in the conduct of probabilistic risk analyses service the nuclear industry.

We all recognize that this new technology for safety analysis of nuclear power plants was invented by NRC research and nurtured through trying times by NRC research. Certainly, all of us here can remember times in which Plenary speakers would not be extolling the virtues of PRA. But, we live in harsh times and the question before us is, now what is research to do for us?

The question is rather remarkable. It is remarkable because the development of PRA as a reliable tool for safety analysis is so far from complete. We really can only do safety analyses using PRA for plants under power operations. What analyses have been done suggest that the public risk from nuclear power plants comes also from low power and shutdown operations. Risks due to fires in nuclear power plants may be commensurate with risks from other causes. But there are not now facile ways to extend the techniques of PRA from power operations to operations under shutdown conditions. Senior Reactor Analysts in the regions do not have tools that allow them to independently validate shutdown plans of licensees. Fire risk assessment seems to have ossified at a level it attained about 15 years ago. Why, then, do we need to ask the question of what shall we do with research? It would seem important to press research to finish the job it has started with the development of PRA so we can apply the findings to low power and shutdown operations, analysis of fire protection, and plant decommissioning. We will want the power of risk assessment technologies to assist in the extension of plant licenses.

Some will say that research does not need to finish the job. Neither the licensees nor the line organizations of NRC may think that they need to have available the complete technology of PRA to provide assurances to the public of safety. I'm certain that the same things were said during the early days of the development of PRA. It was not needed, but it surely is proving useful to have. I'm not so anxious to relegate NRC research to the role of a contractor for line organizations within the NRC.

Though PRA is useful technology, it is not clear that it is widely used technology. Line organizations within NRC are not now bolstering their safety evaluation reports with detailed, plant-specific, probabilistic risk assessments. Regulatory analyses are not laden with detailed risk assessments. So, it appears that NRC research needs to make PRA technology available not just to the specialists but also to the line organizations. Why shouldn't project managers in the future NRC be able to do detailed probabilistic risk assessments of particular plants in response to particular regulatory issues?

NRC research, like many organizations, functions best when it has a clearly defined mission. It has worked best in the past when it had a mission not just to optimize the regulatory structure but also to re-invent the regulatory structure. Many seem to want to relegate NRC research to a limited role of optimizing the current status of regulation. The work of the research organization would be governed then by user needs that so often are limited to the immediate. I am not certain the time is right for so limited a role. We still have much to learn about the PRA technology. We still have to invent ways to apply the technology. Do we want only limited analyses of power plant operations? Do we want in fact to maintain a regulatory structure where goals are set in terms of risk, but analyses are limited to core damage frequency? Can we harmonize a defense in depth regulatory philosophy formulated in a previous era with the risk analysis tools we have now and will have in the future?

Does research sponsored by the regulated industry have an effect on answers to these questions? It must. Assuredly, we see more and more NRC line organizations insisting that licensees will have to provide the technical support and research results for particular licensing actions. But, to know the answer to this question in general, we will have to have some agreement on when NRC will simply review submittals by licensees and when it will independently verify the propositions in the submittals. The answer to this question should affect the future size and the diversity of the NRC research organization. This may well be a question that deserves consideration in future Water Reactor Safety Information Meetings. And, with that, I will close my opening comments.

27th Water Reactor Safety Meeting

How Best to Focus Both on
Maintaining Safety and Reducing Unnecessary Regulatory Burden

Roy P. Zimmerman, Deputy Director
Office of Nuclear Reactor Regulation

Forces Influencing Transition

- Maturing industry
- Improved plant performance
- Improved regulatory tools
- Economic deregulation of electric utilities
- Improved NRC Planning, Budgeting, and Performance Management (PBPM) process

NRC Performance Goals

- Maintain Safety
- Reduce Unnecessary Regulatory Burden
- Increase Public Confidence
- Improve Efficiency/Effectiveness and Enhance Realism of Key NRC Processes

Measures of Success

- Consider performance goals in daily activities
- Practical and useable
- Accepted by Stakeholders
  - Scrutable
  - Predictable
  - Understandable

Inspection & Assessment

- Risk Insights Used to Define Scope and Depth of Inspection Program
- Cornerstones of Assessment Program Derived From Contributors to Plant Risk (i.e., initiating events, mitigation, barriers, emergency planning)
- Specific Inspection Findings evaluated for safety significance using risk insights

Inspection/Assessment Regulatory Framework

NRC's Safety Mission
- Public Health & Safety

Strategic Performance Areas
- Reactor Safety
- Radiation Safety
- Safeguards

Cornerstones
- Reactor Safety: Initiating Events; Mitigation Systems; Barrier Integrity; Emergency Preparedness
- Radiation Safety: Public; Occupational
- Safeguards: Physical Protection

-----Human Performance --------- Safety Conscious Work Environment -------- Problem Identification And Resolution

- Performance Indicators
- Inspection Findings
- Other Information Sources
- Decision Thresholds

Risk-Informed Regulation

- Significant progress made in risk-informing NRC activities with stakeholder involvement (i.e., staff training, improving guidance and developing improved PRA methods and tools
- Pursuing risk-informed modifications to 10 CFR 50 (SECY 98-300)
- Regulatory Guides, Topical Reports and/or pilot plant applications approved in the following areas
  - Inservice Inspection (WOG topical, Vermont Yankee, Surry, ANO)
  - Inservice Testing (Comanche Peak, staff evaluating lessons learned)

Risk-Informing Regulation (Continued)

- Graded Quality Assurance (South Texas, staff addressing barriers to full implementation)
- Technical Specifications (Allowed Outage Time Extensions)
- Other Licensing Initiatives (BWR Vessel Shell Weld Inspections, ANO hydrogen monitoring order, San Onofre hydrogen recombiner exemption/amendment)

- Staff guidance and training (PRA steering committee and risk-informed licensing panel)
- Use of regional senior reactor analysts to evaluate oversight process and licensee events

Examples of ongoing applications -
Balancing safety & unnecessary regualtory burden

- Risk informing decommissioning requirements
- Steam generator tube integrity requirements
- Maintenance rule
- Alternate source term

Risk Informing Part 50 (RIP 50)

- Rulemaking Plan and Advanced Notice of Proposed Rulemaking Made Public on Risk-Informing Special Treatment Regulations
- Integrated decision-making process that uses risk and traditional engineering insights
- Requirements for risk ranking of SSCs, performance monitoring, and other factors needed to maintain risk at accepted levels

Diagram of Categorization and Treatment

Risk Informed		
	(1) RISC-1 SSCs Safety-Related Safety Significant Special Treatment + 50.69 Requirements	(2) RISC-2 SSCs Nonsafety-Related Safety Significant 50.69 Requirements
	(3) RISC-3 SSCs Safety-Related Low Safety Significant Maintain Function	(4) RISC-4 SSCs Nonsafety-Related Low Safety Significant Commercial Treatment

Deterministic

Summary

- Consider Performance Goals
  - Maintaining Safety
  - Reducing Unnecessary Regulatory Burden
  - Improving Effectiveness, Efficiency, and Realism
  - Increasing Public Confidence
- Integration of risk insights with existing design requirements
- Involvement of NRC stakeholders

System Reliability Studies

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ABSTRACT

The U. S. Nuclear Regulatory Commission's Office of Nuclear Regulatory Research analyzes operational data to assess the performance of risk significant systems, structures, and components. Safety system reliability studies are part of these periodic analyses. This paper discusses the results of eight safety system reliability studies: High-Pressure Coolant Injection, High-Pressure Core Spray, Reactor Core Isolation Cooling, Isolation Condenser, Auxiliary Feedwater, Emergency Diesel Generator, Westinghouse Reactor Protection System, and General Electric Reactor Protection System. The results of the studies are presented with associated trends over time.

INTRODUCTION

The U. S. Nuclear Regulatory Commission (NRC) has undertaken an effort to ensure that its policy to expand the use of probabilistic risk assessment (PRA) within the agency is implemented in a consistent and predictable manner. As part of this effort, several studies were performed to evaluate the functional reliability of risk-important safety systems in commercial nuclear power plants.

This paper summarizes the results of studies that were performed to determine the reliability of eight risk-important safety systems over several years of operation: High-Pressure Coolant Injection (HPCI), High-Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC), Isolation Condenser (IC), Auxiliary Feedwater (AFW), Emergency Diesel Generator (EDG), Westinghouse Reactor Protection System (W RPS), and General Electric Reactor Protection System (GE RPS).

The methods used for each study were to: (1) obtain system unreliability estimates based on operating experience data and to compare these estimates with the assumptions, models, and data used in PRAs and individual plant examinations (IPEs); and (2) review the operational data from an engineering perspective to determine trends and patterns and provide insights into the failures and failure mechanisms associated with the operation of the system. Each study analyzed operating experience data contained in Licensee Event Reports (LERs). The Sequence Coding and Search System (SCSS) database was used to identify LERs for classification. Operational data supplied by the Institute of Nuclear Power Operations (INPO), via its Nuclear Plant Reliability Data System (NPRDS) database, was also used to supplement the LER data for the reactor protection systems. NPRDS data was also part of the common-cause failure (CCF) database which was used for estimating CCF parameters in the analyses.

These analyses of operational data provide information on relevant operating experience that can be used to enhance plant inspections of risk-important systems. These analyses can also be used by the NRC staff to perform technical reviews of proposed license amendments, including risk-informed applications. They will be used as input to the Standardized Plant Analysis Risk (SPAR) models for assessing the risk-significance of operational events and conditions in the Accident Sequence Precursor (ASP) program and the new reactor oversight significance determination process. Finally, these analyses will also be used in the development of risk-based performance indicators (RBPIs).

HIGH-PRESSURE COOLANT INJECTION

The HPCI system is a single-train system designed to provide injection of high-pressure coolant when there is a loss of coolant accident (LOCA). It can also provide injection to maintain inventory during events where the reactor is isolated from the main condenser. The analyses of operational experience covered data during 1987-1993 for the 23 U.S. commercial boiling-water reactors (BWRs) that have a dedicated HPCI system (Reference 1).

The HPCI system unreliabilities were estimated using a simplified fault tree model to associate event occurrences with broadly defined failure modes such as failure to start or failure to run. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Several plant risk reports (i.e., PRAs, IPEs, and NUREG reports) were used for comparison with the HPCI reliability results calculated based on the operating experience.

The engineering analysis of HPCI operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the HPCI system:

- The average unreliability of the HPCI system, taking credit for recovery actions, is 0.056. This result assumes that the system is demanded to inject only once during a mission. If, instead, the normally closed injection motor-operated valve (MOV) between the HPCI pump discharge and the RPV is required to open a second time, the unreliability including recovery increases to 0.24. Although observed in the operational data, most PRA/IPEs do not model injection valve cycling. The mean plant-specific unreliability on a single injection, taking credit for recovery, ranged from 0.050 to 0.067.

- The mean plant-specific unreliabilities for a single injection without taking credit for recovery actions are consistent with the values used in 12 of 23 PRA/IPEs. Ten of the other 11 plants had mean unreliabilities greater than a factor of 3 higher than, and outside the uncertainty bounds of, the plant-specific PRA/IPE unreliabilities. The one remaining plant had insufficient information in the PRA/IPE to allow for a comparison. The plant-specific basic event probabilities that account for these differences include:

    - The observed failure-to-run probability is greater than 10 times higher than that used in 13 of the PRA/IPEs.

- The observed failure-to-start probability was in general agreement with the PRA/IPEs. However, two plants had probabilities greater than 6 times higher than those used in the PRA/IPEs.

- The observed failure probability of the injection valve to open on the initial system demand to restore RPV level is more than 10 times higher than that used in 10 of the PRA/IPEs.

- The probability of being out of service for maintenance and testing during a demand to function is in agreement with the PRA/IPEs.

- No correlation was seen between the plant's low-power license date and either the unreliability per operational year or the rate of failures per operational year.

- While the rate of HPCI system unplanned demands and failures per plant operational year decreased during the 7-year period, the associated unreliability showed no significant trend.

- The component failures and their failure mechanisms observed during unplanned demands were different than those found during the performance of surveillance tests.

HIGH-PRESSURE CORE SPRAY

The design function of the HPCS system is to maintain reactor vessel inventory for line breaks up to 1-inch nominal size. The HPCS system also provides spray cooling heat transfer during breaks in which uncovering of the core is assumed. It can also provide injection to maintain the inventory during events where the reactor is isolated from the main condenser. The analyses of operational experience covered data during 1987-1993 for the eight U.S. commercial BWRs that have a HPCS system (Reference 2).

The HPCS system unreliabilities were estimated using a simplified fault tree model to associate event occurrences with broadly defined failure modes such as failure to start or failure to run. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Seven plant risk reports (i.e., PRAs, IPEs, and NUREG reports) were used for comparison with the HPCS reliability results calculated based on the operating experience.

The engineering analysis of HPCS operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the HPCS system:

- The HPCS system operational unreliability (including recovery) estimate calculated from the 1987-1993 experience is 0.075. If recovery is ignored, the operational unreliability estimate is unaffected, since none of the observed failures were recovered. Maintenance-out-of-service is the leading contributor (67%) to HPCS operational unreliability followed by failure of the injection valve (27%).

- The HPCS system mean unreliability estimates approximated from the PRA/IPE data are lower than the mean estimates derived from the 1987-1993 experience. The reasons for this difference appear to be the lower failure probabilities used in the PRA/IPEs for the maintenance out of service and failure to run of the HPCS injection subsystem. However, the pump train failure to run rate (1987-1993 experience) was based on sparse data, with no failures in 316 hours. Further, the HPCS motor run times were short. Additional data (i.e., operating experience) are needed before high confidence can be placed on either the PRA/IPE failure to run estimate or the estimate based on 1987-1993 experience.

- No trends were identified in the HPCS operational unreliability when plotted against low-power license date or when plotted with regard to calendar year.

REACTOR CORE ISOLATION COOLING

The RCIC system is a single train standby system designed to ensure that sufficient reactor water inventory is maintained in the vessel following isolation from the normal heat sink. The analyses of operational experience covered data during 1987-1993 for the 29 U.S. commercial BWRs that have a RCIC system (Reference 3).

The RCIC system unreliabilities were estimated using a simplified fault tree model to associate event occurrences with broadly defined failure modes such as failure to start or failure to run. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Twenty-one plant risk source reports (i.e., PRAs, IPEs and NUREG reports) were used for comparison with the RCIC reliability results obtained in this study.

The engineering analysis of RCIC operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the RCIC system:

- The RCIC system unreliability (including recovery) calculated based on the operating experience data in which RCIC is required to inject to the reactor vessel for short term missions (less than 15 minutes) is 0.04. The short term missions typically follow a reactor scram where feedwater is available and the main steam isolation valves are open. If recovery is excluded, the short term mission unreliability is 0.06. This unreliability is primarily attributed to failures to start, typically as a result of problems in controlling turbine speed where the problem is caused by either personnel error or hardware problems that result in turbine overspeed trips.

- The estimate of RCIC system unreliability calculated based on the operating experience data in which RCIC is required to inject to the reactor vessel for missions that are longer than 15 minutes and up to several hours is 0.08. The long term missions typically follow a reactor scram where feedwater is not available and/or the reactor vessel is isolated. If recovery is excluded, the long term mission unreliability is 0.16. The difference in the unreliability estimate calculated for the long term missions as compared to the short term missions is attributed mainly to restarting the turbine to maintain reactor vessel water level. This unreliability is primarily due to hardware failures associated with restarting the turbine or the cycling of motor-operated valves.

- Comparing the estimates of RCIC system unreliability calculated from the information contained in PRA/IPEs to the estimates (with recovery) calculated from the operating experience data revealed that most (approximately 75%) of the PRA/IPE point estimates lie within the uncertainty interval associated with the operating experience estimate. However, about 21% of the PRA/IPE estimates predict better performance than identified by the estimates calculated from the operating experience data.

- Most of the PRA/IPEs do not model the RCIC system in the way it is observed to be operated in the operating experience data. Specifically, the maintenance of reactor vessel water level by either restart and/or recirculation following initial injection is generally not modeled. For the PRA/IPEs that model the system with the restart and/or recirculation modes of RCIC, the failure probabilities assigned to these modes of operation appear to be optimistic. For example, the initial failure to start (other than the injection valve) probabilities and the restart failure probabilities differ by about a factor of 2.6 from the operating experience data. However, the PRA/IPEs use the same probabilities for restart as for initial start. According to the operating experience data, the failure to restart contribution to overall unreliability is about a factor of two greater than the failure to start (other than the injection valve) contribution (27% versus 12%, respectively).

- The operating data contained five instances where multiple systems (RCIC, HPCI, and sometimes reactor water cleanup) either had failed or had the potential to fail concurrently; these instances may be common cause failures. The events involved motor-operated valves, the steam leak detection circuitry, and the turbine governors. In two of the five instances the RCIC and HPCI systems were affected during an unplanned demand. The other events were discovered during surveillance testing (2) and other routine plant operations (1).

- For the short term missions, a decreasing trend in RCIC system unreliability with respect to calendar year was identified by statistical analysis of the operating data. In addition, some indication of a trend was identified in the short term unreliability with regard to low-power license date, but it is not a strong indication. No statistical trends were identified with regard to long term RCIC unreliability.

- The unplanned demand frequency exhibits a statistically significant decreasing trend. This is likely a result of a corresponding decrease in unplanned plant trips, which typically include a RCIC system actuation.

- Failure frequency exhibits no trend when plotted against plant operating year. There was no correlation observed between the plant's low-power license date and the frequency of failures per operating year.

ISOLATION CONDENSER

The IC system is a standby high-pressure system that removes residual and decay heat from the reactor vessel in the event of a scram in which the reactor becomes isolated from the main condenser, or if any other high pressure condition exists. The analyses of operational experience covered data during 1987-1993 for the five U.S. BWRs that have an IC system (Reference 4).

The IC system unreliabilities were estimated using a simplified fault tree model to associate fault event occurrences with broadly defined failure modes such as failure to operate or failure to provide makeup. The failure probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each failure by failure mode and then estimating the corresponding number of demands (both success and failures). IC train and component failure rates were also extracted from PRA/IPEs. These were then combined, consistent with the quantification performed using the operating experience data. The resulting failure mode probabilities were then compared to the system level unreliability estimates and failure mode probabilities calculated for this study.

The engineering analysis of IC operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the IC system:

- The IC train unreliability (including recovery), based on operational experience data, is 0.02. The failure to operate failure-mode (i.e., failure to achieve stable steam flow from the reactor to the condenser, and condensate flow from the condenser to the reactor) and failure to provide makeup failure-mode (i.e., failure to provide makeup water to the shell side of the isolation condenser) contributed equally to the overall unreliability. The recovered and non-recovered train unreliability estimates differ by a factor of five. The difference is primarily attributable to the spurious isolations of the IC train as observed in the unplanned demands. All the failures observed for the IC train failing to operate were caused by spurious isolation of the IC train.

- The average of the estimates of IC train unreliability based on information contained in the PRA/IPEs was generally about a factor of 1.5 lower than the estimate of the mean probability based on operational experience data. All of the PRA/IPE estimates of IC train unreliability are within the uncertainty interval based on the operational experience data. However, the PRA/IPEs show that the condensate isolation valve failing to open is the dominant contributor to IC train unavailability. This contrasts with the calculations based on operational data, which show that the effect of this type of failure was not as important to IC train unreliability as the spurious isolations of the IC train.

- No statistically significant trends in IC train failure and unplanned demand frequencies or unreliability by calendar year were observed in the operational experience data. Further, IC train unreliability was analyzed against low-power license date and no trends were observed.

AUXILIARY FEEDWATER

The AFW system is designed to provide feedwater to the steam generators to maintain a heat sink in the event of (1) a loss of main feedwater, (2) a reactor trip and loss of offsite power, and (3) a small break loss of coolant accident. The analyses of operational experience covered data during 1987-1995 for the 72 U.S. pressurized water reactors (PWRs) that have an AFW system (Reference 5).

The AFW system unreliabilities were estimated using a simplified fault tree model to associate event occurrences with broadly defined failure modes such as failure to start or failure to run. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Forty-seven plant risk reports (i.e., PRAs, IPEs, and NUREG reports) were used for comparison with the AFW reliability results calculated based on the operating experience.

The engineering analysis of AFW operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the AFW system:

- During 1987-1995 there were no failures of the entire AFW system identified in 1,117 unplanned system demands. Using a system level fault tree model that combines individual failure modes, the average operational unreliability of the AFW system is 3.4E-05. Individual plant results vary over two orders of magnitude, from 1.5E-06 to 6.2E-04. The variability largely reflects the diversity found in AFW system designs. However, there is some variation in results among plants with similar AFW designs. This is attributed to the plant-to-plant differences in the 1987-1995 experience data, and to a lesser degree, differences in the levels of redundancy in the feed control/injection headers.

- AFW designs composed of only turbine-driven pumps were the least reliable, while AFW designs comprising three redundant trains of diverse design (e.g., two motor and one turbine driven pumps) were more reliable. AFW designs consisting of four trains (three motor and one turbine) are not significantly different in reliability terms from the three train (two motor and one turbine) pump designs. The benefits of additional trains of redundancy to AFW system reliability is offset by the effects of common cause failures. Although the AFW designs consisting solely of turbine-driven pumps tend to be less reliable in routine operations, for potential station blackout situations, they would be more reliable than their counterparts with multiple motor-driven pump trains.

- Some differences were found between the AFW reliability estimated from the operating experience data and that estimated from PRA/IPE data. The major differences between the two estimates are attributable to the probabilities associated with failure of the primary AFW system water source (e.g., CST suction path, generally not considered as being probabilistically important in most PRA/IPEs), and the AFW turbine-driven pump failure to run (a significantly higher failure rate results when using the relatively limited 1987-1995 experience data).

- The loss of suction source was a dominant contributor to many of the design classes. This event, though rare, is important because it disables the designed redundancy of the AFW systems and is usually discounted or not modeled in PRAs. There was one failure of a suction source during the 1,117 unplanned system demands observed in the operational experience. This failure occurred during an automatic start of two motor-driven pumps. Suction pressure was insufficient for pump operation and caused an automatic shift to the alternate source (service water). The low suction pressure condition was a result of operating with the condensate storage tank isolated, while not maintaining adequate level in the upper surge tank. Even though AFW pump suction shifted to the alternate source (service water), the service water system was fouled with clams and sludge which caused the AFW flow control valves to the steam generators to clog and significantly reduce flow to two of four steam generators.

- No trends were identified in the AFW operational mission unreliability when plotted against calendar year or low-power license date. The unplanned demand frequency as a function of calendar years exhibited a statistically significant decreasing trend. When unplanned demand frequency is plotted against low-power license dates, a statistically significant increasing trend was identified which appears to be related to an increasing trend in automatic reactor trips.

EMERGENCY DIESEL GENERATOR

The EDG train is part of the standby emergency onsite ac power system and is required to be available as a reliable source of ac power in the event of a loss of normal ac power during all plant modes (operating or shutdown). The EDG train is designed to have sufficient capacity to power all the loads required to safely shut the plant down or supply emergency core cooling system (ECCS) loads during a loss-of-coolant accident (LOCA). The analyses of operational experience covered data during 1987-1993 for all U.S. nuclear power plants (Reference 6).

Approximately 60% of the plants are required to report EDG train failures in accordance with requirements in Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electrical Power Systems." The analyses focused on these plants.

EDG train unreliabilities were estimated using a simplified fault tree model to combine broadly defined train failure modes such as failure to start or failure to run into an overall EDG train unreliability. The failure probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each failure event by failure-mode, and then estimating the corresponding number of demands (both successes and failures). Approximate PRA/IPE-based unreliabilities were calculated from the failure data documented in the respective PRA/IPE for the start, load, run, and maintenance phases of the EDG train operation.

The engineering analysis of EDG operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the EDG train:

- The estimated EDG train unreliability derived from unplanned and cyclic test demand data for the RG-1.108 plants was 0.044. The unreliability estimate includes consideration of recovery of EDG train failures, maintenance out of service while the plant is not in a shutdown condition, and assumes an 8-hour mission time. Maintenance out of service is the major contributor to EDG train unreliability. Approximately 70% of the unreliability is attributed to maintenance being performed on an EDG train at the time of an unplanned demand. If recovery is excluded, the estimate of an EDG train unreliability is 0.069.

- The average of the plant-specific RG-1.108-based estimates of EDG train unreliability is in general agreement with the average of the PRA/IPE estimates, assuming an 8-hour run time of the EDG. Generally, the RG-1.108-based estimate for failure to start and maintenance out of service probabilities agree with their respective PRA/IPE counterparts. However, for a 24-hour mission time for the EDG train, the average PRA/IPE estimate of failure to run is approximately a factor of 30 higher than the corresponding RG-1.108-based estimate. This is because the analysis found a lower failure-to-run rate based on cyclic-test 24-hour run data than was used in PRAs.

- Based on the mean reliability, all of the RG-1.108 plants (44) with an EDG target reliability goal of 0.95 attain the SBO target goal provided that the unavailability of the EDG due to maintenance is ignored. For the RG-1.108 plants with a EDG target reliability goal of 0.975, eighteen of the nineteen RG-1.108 plants, based on the mean reliability, attain the reliability goal provided that the unavailability of the EDG due to maintenance is ignored. The EDGs associated with the plant not achieving the 0.975 reliability goal had a mean reliability of 0.971.

- The effect of maintenance unavailability on EDG reliability is significant based on the RG-1.108 data. The technical basis for the Station Blackout Rule assumes that such unavailability is negligible (0.007). The estimate derived from the RG-1.108 data for maintenance out of service is 0.03. If maintenance out of service is included in the reliability comparison, forty of the 44 RG-1.108 plants with a 0.95 target reliability attain the goal. None of the EDGs with a 0.97 target reliability meet the reliability goal.

- Trending analysis of the failure rate, unplanned demand rate and unreliability data by calendar year indicates no statistically significant trend over the 7 years of the study period. The analysis of plant-specific unreliability by low-power license date indicates no statistically significant trend. However, analysis of the number of plant-specific EDG failures per year as a function of low-power license date identifies a statistically significant trend. The trend indicates that the plants with low-power license dates from 1980-1990 typically had more EDG failures during the 7-year period than those plants with a low-power license date prior to 1980. Information in the LERs was not sufficient to determine the reason for the trend.

WESTINGHOUSE REACTOR PROTECTION SYSTEM

The Westinghouse RPS is a control system for rapidly shutting down the nuclear reactor by inserting control rods. It consists of control rods, trip breakers, control logic, and instrumentation. All Westinghouse RPS designs have similar control rods and trip breakers. However, the control logic used in approximately 70% of the RPS designs is the Solid State Protection System (SSPS) and the remaining 30% use analog logic. Approximately 85% of the RPS designs have analog instrumentation systems (Analog Series 7300 or earlier models) and the remaining 15% have the Eagle-21 solid state system. The analyses of operational experience covered data during 1984-1995 for the Westinghouse RPS (Reference 7).

The Westinghouse RPS unreliabilities were estimated using a simplified fault tree model to associate inoperabilities and unplanned actuations with component failures. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Several plant risk reports (i.e., PRAs, IPEs, and NUREG reports) were used for comparison with the Westinghouse RPS reliability results calculated based on the operating experience.

The engineering analysis of Westinghouse RPS operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the Westinghouse RPS:

- Based on the operating experience, the mean unavailability (failure probability upon demand) was 2.2E-5 (with no credit for manual scram by the operator) for the Analog Series 7300 design. The lower 5th percentile is 5.8E-6 and the upper 95th percentile is 5.7E-5. Approximately 95% of the overall RPS unavailability is from CCF events. CCF of the two undervoltage driver cards (one per train) is the dominant contributor (46.1%) to RPS unavailability. Other important CCF events involve the channel bistables (11.5%), train universal cards (9.7%), channel signal processing modules (7.8%), reactor trip breakers (7.4%), and rods (5.5%). Results for the Eagle-21 RPS design are similar, with a mean unavailability of 2.0E-5. The Analog Series 7300 and Eagle-21 designs have comparable unavailabilities because their common features are the dominant contributors.

- Both the Analog Series 7300 and Eagle-21 RPS designs have a single undervoltage driver card in each of the two trains. Failure of both of these cards results in failure of RPS (unless manual scram is credited). This CCF event is the dominant contributor (almost 50%) to RPS unavailability. In 1989, a CCF event involving both driver cards occurred while a plant was shut down. The failures were caused by maintenance activities and were detected before the plant returned to power. Since then, the driver card design has been changed to minimize the chance of such maintenance activities causing such failures. Also, plant procedures for such maintenance have been improved. However, CCF of both of these cards is still predicted to be a dominant contributor to RPS unavailability.

- Issues related to reactor trip breakers, arising during the early 1980s, are no longer dominant with respect to RPS unavailability. Automatic actuation of the shunt trip mechanism within the reactor trip breakers and improved maintenance procedures have resulted in improved performance of these components.

- If credit is given for manual scram, the mean unavailabilities are 5.5E-6 for the Analog Series 7300 design and 4.5E-6 for the Eagle-21 design. Operator action reduces the RPS unavailability by approximately 75%. This reduction is significant and occurs mainly because the manual scram signal bypasses the dominant undervoltage driver card failures.

- RPS unavailability estimates from IPEs and other sources range from approximately 1.0E-6 to 1.0E-4. Because of the lack of detailed information in the IPE submittals, it is not clear which estimates included credit for operator action. The IPE range of RPS unavailabilities covers the uncertainty ranges obtained in this study, based on the analysis of data from 1984 through 1995. However, most of these other sources estimated that the trip breaker CCF events would dominate the RPS unavailability. In this study such events contribute less than 10% when no credit is taken for manual scram by the operator, and approximately 30% if credit is taken.

- The engineering analysis identified decreasing trends in component failure and CCF event counts for several RPS components. No increasing trends were identified over the period 1984 through 1995.

GENERAL ELECTRIC REACTOR PROTECTION SYSTEM

The General Electric RPS is a control system for rapidly shutting down the nuclear reactor by inserting control rods. It consists of signal channels, trip systems, hydraulic control units (HCUs), and control rods. All General Electric RPS designs have similar signal channels, HCUs, and control rods. All General Electric plants except Clinton have relay-based trip systems; Clinton has a solid-state trip system. The analyses of operational experience covered data during 1984-1995 for the General Electric RPS (Reference 8).

The General Electric RPS unreliabilities were estimated using a simplified fault tree model to associate inoperabilities and unplanned actuations with component failures. The probabilities for the individual failure modes were calculated by reviewing the failure information, categorizing each event by failure mode, and then estimating the corresponding number of demands (both successes and failures). Several plant risk reports (i.e., PRAs, IPEs, and NUREG reports) were used for comparison with the General Electric RPS reliability results calculated based on the operating experience.

The engineering analysis of General Electric RPS operational data investigated trends and patterns in system failures and demands based on operational time, low-power license date, subsystem, cause, and method of discovery.

The following is a summary of the major findings for the General Electric RPS:

- Based on operating experience, the mean unavailability (failure probability upon demand) of 5.8E-6 for the BWR/4 relay-based design. This unavailability does not include any credit for operator action to actuate the manual scram switches. An uncertainty analysis resulted in a 5th percentile value of 1.8E-6 and a 95th percentile value of 1.4E-5. Essentially 100% of this unavailability is from CCF events; the combinations of independent failures contribute less than 0.1%. Channel failures contribute 58% to the total unavailability, hydraulic control unit failures contribute 32%, trip system failures contribute 6%, and control rod and control rod drive failures contribute 4%.

- CCF events involving the scram pilot solenoid-operated valves (SOVs) and backup scram SOVs contribute 29% to the overall RPS unavailability. The most significant operational event, involving the use of improper seating material and affecting all the scram pilot SOVs, occurred in 1984. Two similar types of SOV CCF events occurred in 1994 but did not affect as many of the components. Also, problems with the use of liquid thread sealant resulted in several significant CCF events. The requirement to test 10% of the control rods every four months helped uncover these types of problems (developing over time) before they developed to catastrophic failures.

- The backup scram portion of the RPS may be an important contributor to low RPS unavailability. (Without the backup scram logic, only two of eight trip system relay failures are needed to fail the RPS, rather than four of eight if the backup scram system is modeled.) The backup scram SOVs are classified as non-safety-related, and these valves are not part of the NPRDS reportable scope for the General Electric RPS. Therefore, no failure data was available for these valves. Also, it is not clear how often these valves are tested. This study assumed these valves are tested every 18 months during shutdown, and that their failure characteristics are similar to the scram pilot SOVs.

- There were significant scram discharge volume (SDV) problems in the early 1980s involving both drainage of SDVs and level instrumentation, dominated by the 1980 Browns Ferry Unit 3 failure of 76 of 185 control rods to insert. Data collected during the period 1984 through 1995 indicate that SDV instrumentation failure probabilities are similar to other RPS trip instrumentation. Also, only one inadvertent filling of the SDV while a plant was at power was identified during the period. Finally, the RPS fault tree quantification indicates that SDV events leading to failure of the RPS contribute less than 1% to the overall RPS unavailability. Therefore, early SDV-related problems in General Electric RPSs are no longer dominant contributors to RPS unavailability.

- If credit is given for manual scram, the resulting RPS unavailability is 2.6E-6. Operator action reduces the RPS unavailability by approximately 55%. This reduction is limited because a dominant contributor to RPS unavailability is the scram pilot SOV CCF event, which is unaffected by the operator action. Also, the manual scram signal must still pass through the channel and trip system relays, for the configuration analyzed.

- The unavailability estimate of 5.8E-6 (allowing no credit for manual scram by the operator) is lower than typically used in the past PRA/IPEs. Past estimates typically ranged from 1.0E-5 to 3.0E-5 and were usually based on information in NUREG-0460, published in 1978. The individual component failure probabilities per demand, derived from the 1984 through 1995 data, are generally comparable to failure probability estimates listed in previous reports. Therefore, the low RPS unavailability estimate is mostly attributable to lower failure probabilities for the CCF events. The General Electric RPS CCF events collected for this project, covering the period 1984 through 1995, contain few events involving complete failures of many redundant components. Correspondingly, the CCF calculations result in low CCF failure probabilities.

- The trends in component failure probabilities and numbers of CCF events are generally flat over the period 1984 through 1995.

SUMMARY

The results of each study are summarized in the table below, where a downward-sloping arrow indicates a decreasing trend and a horizontal arrow indicates no evidence of a trend.

Study	Mean Unreliability	Unplanned Demand Trend	Failure Rate Trend	Unreliability Trend	Consistency with PRA/IPEs	Unreliability vs Plant Age Trend
HPCI (1987-1993)	0.06	↘	↘	⇨	General agreement--few plants lower than operating experience	None
EDG--RG1.108 (1987-1993)	0.04	↘	↘	⇨	General agreement--fail-to-run higher in PRAs	None
IC (1987-1993)	0.02	⇨	⇨	⇨	General agreement--nature of failures differ	None
RCIC (1987-1993) Short (<15 min) Long (>15 min)	 0.04 0.2	↘	⇨	↘	General agreement--restart different in PRAs	None
HPCS (1987-1993)	0.08	↘	⇨	⇨	Fail-to-run contribution higher in operating experience	None
AFW (1987-1995)	3E-5	↘	⇨	⇨	Fail-to-run and suction contributions higher in operating experience	None
W RPS (1984-1995)	2E-5	↘	N/A	N/A	General agreement--reactor trip breaker contribution different	N/A
GE RPS (1984-1995)	6E-6	↘	N/A	N/A	One order of magnitude lower than IPEs	N/A

REFERENCES

1. Reliability Study: High-Pressure Coolant Injection System, 1987-1993 (NUREG/CR-5500, Vol. 4)

2. Reliability Study: High-Pressure Core Spray System, 1987-1993 (NUREG/CR-5500, Vol. 8)

3. Reliability Study: Reactor Core Isolation Cooling System, 1987-1993 (NUREG/CR-5500, Vol. 7)

4. Reliability Study: Isolation Condenser System, 1987-1993 (NUREG/CR-5500, Vol. 6)

5. Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995 (NUREG/CR-5500, Vol. 1)

6. Reliability Study: Emergency Diesel Generator Power System, 1987-1993 (NUREG/CR-5500, Vol. 5)

7. Reliability Study: Westinghouse Reactor Protection System, 1984-1995 (NUREG/CR-5500, Vol. 2)

8. Reliability Study: General Electric Reactor Protection System, 1984-1995 (NUREG/CR-5500, Vol. 3)

Development of Risk-Based Performance Indicators

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ABSTRACT

The U. S. Nuclear Regulatory Commission (USNRC) is moving toward more risk-informed performance-based oversight and regulation. The new oversight process uses a combination of performance indicators and risk-informed inspections. The development of risk-based performance indicators (RBPIs) is a planned future improvement to this process. This paper documents the activities for developing RBPIs within the framework of the new oversight process.

I. INTRODUCTION

One mission of the USNRC is to ensure that operation of U. S. nuclear power plants provides adequate protection of public health and safety (Ref. 1). The USNRC has adopted the policy of increasing the use of probabilistic risk analysis (PRA) methods in the regulatory process (Ref. 2). Risk methods provide a framework for integrating aspects of performance on a common quantitative (and qualitative) basis. SECY-99-007 (Ref. 3) documents the new risk-informed reactor oversight process and how it ensures that the USNRC mission is met through the use of a combination of performance indicators and risk-informed inspection activities.

As shown in Figure 1, the new oversight process uses a hierarchical regulatory oversight framework where performance affecting "Cornerstones of Safety" (cornerstones) is monitored through a combination of performance indicators and risk-informed inspections. These inspections verify and validate the performance indicator data and supplement that data in areas where performance indicators are limited or, as of now, not developed or defined. Both the performance indicators and the risk-informed inspection activities are objectively evaluated against risk-informed performance thresholds to the extent currently practical. When licensee performance is outside the normal performance band, or is in further decline, a graded inspection response and other diagnostic and corrective regulatory actions are taken.

The majority of the performance indicators and associated thresholds presently defined for the oversight process are count-driven and several have limited direct relationship with risk. This paper discusses the efforts to develop RBPIs which are planned as a future improvement to the oversight process.

Risk-based performance indicators are intended to provide objective measures for use in assessing reactor operational performance in the context of public risk. They relate plant performance to the cornerstones either directly or through the integration of the constituent parts that contribute to the risk. The RBPIs include performance metrics such as frequency, reliability, availability, and probability that are measurable, calculable and directly relate to public risk through accident sequence logic. The RBPIs provide complementary and more comprehensive risk information on plant performance both at an industry level and at a plant level than the current oversight process indicators. Likewise, the thresholds for the RBPIs provide the basis for judging the significance of performance changes directly associated with corresponding changes in public risk. Their use in the reactor oversight process ties regulatory actions more closely to risk performance.

II. SUPPORTING PROGRAMS

The RBPIs are the culmination of programs which the USNRC and industry have undertaken over the past several years. Figure 2 shows the data and analyses programs that have preceded and currently support the development of RBPIs. The figure is divided into four tiers of "Operational Data," "Industry-Wide Analyses," "Plant-Specific Event Analyses," and "RBPI Development." The dotted lines indicate activities under development while the solid lines are completed or fully functional efforts.

Operational experience data ("Operational Data" in Fig. 2) on plant performance are the source of raw performance information used in this process. The Institute of Nuclear Power Operations is providing the primary source of equipment performance data which is essential for the production of risk-related studies and RBPIs through the Equipment Performance and Information Exchange (EPIX) system, which replaces the Nuclear Plant Reliability Data System. Databases on licensee event reports (Sequence Coding and Search System [SCSS]) and monthly operating reports (MORs) were developed several years ago and are currently available to provide additional needed performance data. The USNRC Reliability and Availability Data System (RADS) is being developed to integrate the necessary information from these various data sources and to provide reliability and availability parameter estimates as a primary input to the RBPIs.

Several risk-related studies ("Industry-Wide Analyses" in Fig. 2) of reactor operating experience have been completed which provide a baseline of performance characteristics, models, and methods from which RBPIs will evolve (Refs. 4 through 13). These studies include analyses of initiating event frequencies, reliabilities of risk-important systems and components, and evaluation of common-cause failure events and parameters. This has included trending industry-wide performance, comparing results with probabilistic risk assessments, individual plant examinations and regulatory issues, and identifying engineering insights for use in regulatory activities such as risk-informed inspections.

Plant-specific evaluations of safety performance ("Plant-Specific Event Analyses" in Fig. 2) rely on the information derived from evaluation of the "Industry-Wide Analyses" mentioned above and plant-specific design and operational features. The Accident Sequence Precursor (ASP) program (Ref. 14) uses the "Industry-Wide Analyses" results in the evaluation of the risk significance of plant-specific operational events using the Simplified Plant Analysis Risk (SPAR) models (Ref. 15). Results from the ASP program are expected to be a part of or provide input to the RBPIs. In addition, the SPAR models provide a framework to integrate risk-based performance information for both performance indicators and inspection findings.

Integrating the methods, models, and data of the current programs to yield a set of RBPIs is the focus of this project. Shown in Figure 2 as the two items in the "RBPI Development" tier, these involve the development of RBPIs through the adoption of the methods, models and data just discussed, and the implementation of a process to routinely collect, analyze and present plant-specific performance results.

III. CONCEPTS

There are two fundamental principles which apply to the development of RBPIs:

- RBPIs must measure performance related to the cornerstones of the reactor oversight process;
- RBPIs must measure performance as it relates to risk.

The RBPIs must have the following attributes to meet the first principle:

- directly relate to the cornerstone goals;
- compatible with and complementary to the risk-informed inspection activities of the oversight process;
- regulatory action thresholds consistent with the new oversight process;
- timely and accurate with minimal misidentification of performance problems;
- cover all modes of plant operation.

The RBPIs must have the following attributes to meet the second principle:

- relate logically to risk and the constituent elements of risk;
- sensitive to risk significant changes in performance;
- cover the systems, structures and components that have been shown to be important contributors to risk;
- amenable to establishing risk-informed thresholds of performance.

These general attributes are expected to have the following technical characteristics that connect them to plant risk:

- relate directly to parameters such as reliability, availability, probability and frequency that are the key constituents of measuring risk and related performance;
- have definitions that are consistent across the industry but are also capable of accounting for differences in plant design and operation that could affect indicator results and associated thresholds selected to detect declining performance. For example, indicators of reliability and availability for emergency diesel generators may be similar but their plant-specific threshold may vary depending on the number of diesel generators at each plant. Similarly, different plant-specific indicators may have similar overall thresholds. For example, auxiliary feedwater system reliability and availability performance indicators may have the same threshold for all plants but the specific plant indicator may be made up from a combination of models of train indicators based on the type of pump (such as turbine, motor or diesel) and the number of injection paths (such as two, three, or four);
- account for the sparseness of available data and associated statistical variance in identifying whether plant specific differences exist from industry or group performance (and whether performance changes can actually be detected). For example, data exists to track the frequency of risk-important initiators such as loss of offsite power or loss of feedwater. However, for events such as a loss-of-coolant accident due to large pipe breaks, the mean time between occurrences is so long (none have occurred yet) that they are not amenable to direct measurement as an RBPI;

- thresholds for determining deviations from expected norms and establishing limits of acceptable performance should be able to distinguish between normal fluctuations (in order to prevent false identification of problems when none exist) and bona fide trends of poor performance (to allow for timely action to correct problems).

It is important to note that risk-informed inspection will provide the primary source of licensee performance information for events of high potential safety consequence but low frequency, such as loss of coolant accidents, steam generator tube ruptures, and seismic events. Agency response for these events will be immediate and not dependent on performance indicator trends and action thresholds. Risk-informed inspections will likewise provide the primary source of licensee performance information in instances where RBPIs cover some, but not all, capability concerns (e.g., design issues related to infrequently occurring conditions or scenarios will be captured as they are discovered). Risk-informed inspections will also address the risk-significant aspects of the factors not covered by the RBPIs.

Figure 3 shows the risk-based hierarchy and the associated levels of indication that will form the bases for the RBPIs. Performance problems at a plant that are risk-significant will be manifested in one or more of the levels in this hierarchy. The levels of the RBPIs in this framework devolve from industry and sequence level indications suitable for comparison to the USNRC Safety Goals. These further devolve to system, train, and basic event level indicators which are constituent parts of plant risk. In this sense, the lower level indicators of Figure 3 are "leading" indicators of overall risk. This framework covers potentially risk significant performance for all areas of operation, including at power or shutdown/refueling events, and internal or external events. The availability of data, methods, and models determine the quality and capability of RBPIs at the various levels in the hierarchy, but the framework is conceptually complete.

IV. DEVELOPMENT PLAN

Using readily available data and existing and peer reviewed methods and models, RBPIs will be developed in phases. Table 1 provides a listing of the RBPIs envisioned for each of these phases.

A. Phase 1

During the current phase (Phase 1), RBPIs will be defined for the cornerstones of initiating events and mitigating systems, both during power operation and shutdown/refueling. An integrated sequence/plant level indicator will also be developed. This integrated risk-based indicator will use SPAR models to integrate cumulative risk implications of multiple indicators and inspection findings. It will use indicator values for initiating event frequency, mitigation system performance, and when available, containment/barriers.

Using data, models and analytical techniques developed in the studies previously referenced, the following characteristics of and analysis methods for the RBPIs listed in Table 1 for the "Initiating Events" and "Mitigating Systems" cornerstones will be investigated:

- data requirements, sources and availability;
- time periods required to identify trends;
- sensitivity to grouping (e.g., peers, vendor, reactor type, trains, etc.);

- sensitivity to plant operating conditions;
- capability of providing leading performance indication;
- ability to be validated;
- statistical trending methods;
- statistical methods to identify baselines of normal variance in industry and plant-specific performance;
- statistical methods to identify adverse performance trends that are outside expected normal variance;
- threshold values corresponding to the reactor oversight process action bands.

Throughout this development process there will be broad internal and public review, including a public workshop. The preliminary and final results will also be discussed in public presentations to the Advisory Committee on Reactor Safeguards (ACRS) and to the Commission. The current schedule is to complete the development and implementation of this set of RBPIs by January 2001.

B. Phase 2

The second phase of the development will determine RBPIs for the cornerstone of barriers, both at power operation and shutdown/refueling. Completion of indicators found during the first phase to be impractical due to the lack of models or data will also be attempted during this phase.

C. Deferred or Not to be Developed

No risk models are presently under development for the cornerstones of emergency preparedness, public radiation safety, occupational radiation safety and physical protection. Consequently, no RBPI development effort is envisioned to be undertaken in these cornerstone areas.

REFERENCES

1. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Strategic Plan", NUREG-1614, Vol. 1., September 1997, NRC: Washington, D.C. (1997).

2. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register*, Vol. 60, p. 42622 (60 FR 42622), August 16, 1995, NRC: Washington, DC (1995)..

3. Nuclear Regulatory Commission (U.S.), Washington, D.C. "Recommendations for Reactor Oversight Process Improvements," SECY-99-007, January 8, 1999, NRC: Washington, DC (1999).

4. G. M. Grant, et al. "High-Pressure Coolant Injection (HPCI) System Performance, 1987-1993," AEOD/S95-02 (INEL-94/0158), February 1995, NRC: Washington, D.C. (1995).

5. G. M. Grant, et al. "Isolation Condenser System Reliability, 1987-1993," AEOD/S96-01 (INEL-95/0478) August 1996, NRC: Washington, D.C. (1996).

6. G. M. Grant, et al. "Emergency Diesel Generator Power System Reliability 1987-1993," AEOD/S96-03 (INEL-95/0035), February 1996, NRC: Washington, D.C. (1996).

7. J. P. Poloski, et al. "Reactor Core Isolation Cooling System Reliability, 1987-1993," AEOD/S97-02 (INEL-95/0196), June 1997, NRC: Washington, D.C. (1997).

8. J. R. Houghton, "Special Study Fire Events - Feedback of U.S. Operating Experience," AEOD/S97-03, June 1997, NRC: Washington DC (1997).

9. J. P. Poloski, G. M. Grant, C. D. Gentillon, "High-Pressure Core Spray System Reliability, 1987-1993," AEOD/S98-02 (INEEL/EXT-95-00133), May 1998, NRC: Washington, D.C. (1998).

10. C. L. Atwood, et al. "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980-1996," NUREG/CR-5496 (INEEL/EXT-97-00887), November 1998, NRC: Washington, D.C. (1998).

11. J. P. Poloski, et al. "Reliability Study: Auxiliary/Emergency Feedwater System, 1987-1995," NUREG/CR-5500, Vol. 1 (INEEL/EXT-97-00740), August 1998, NRC: Washington, D.C. (1998).

12. J. P. Poloski, et al. "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995," NUREG/CR-5750 (INEEL/EXT-98-00401), February 1999, NRC: Washington, D.C. (1996).

13. F. M. Marshall, D. M. Rasmuson, A. Mosleh, "Common-Cause Failure Database and Analysis System: Overview," NUREG/CR-6268 Vol. 1-4 (INEEL/EXT-97-00696), June 1998, NRC: Washington, D.C. (1998).

14. R. J. Belles, et al. "Precursors to Potential Severe Core Damage Accidents," NUREG/CR-4674 (ORNL/NOAC-232, Vols. 1 - 26), NRC: Washington, D.C. (1998).

15. S. M. Long, et al. "Current Status of the SAPHIRE Models for ASP Evaluations," *Probabilistic Safety Assessment and Management-PSAM 4*, New York, NY, Sept. 13-18, 1998, page 1195, Springer-Verlag, London, UK (1998).

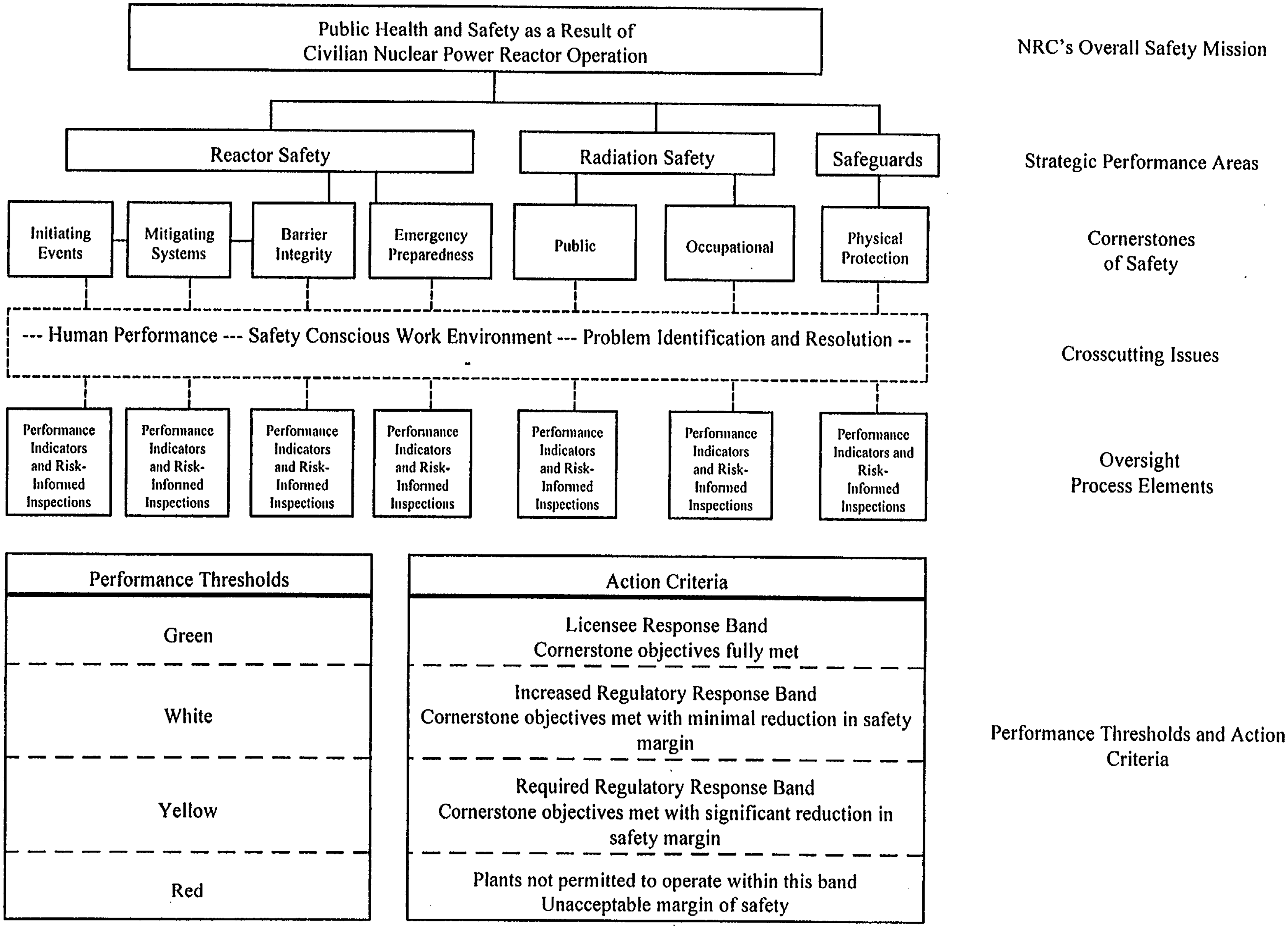

Figure 1 - Reactor Oversight Process

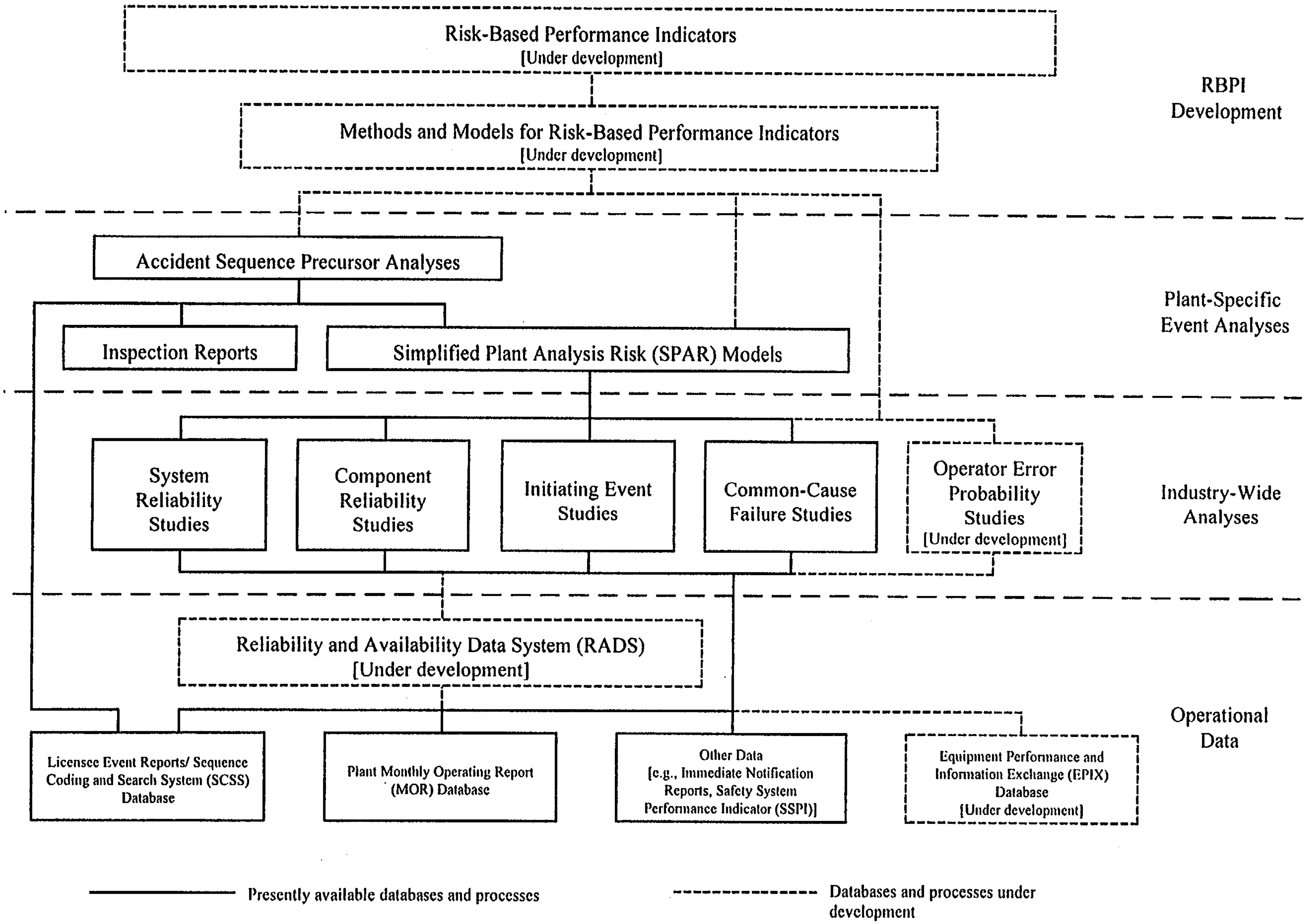

Figure 2 - Programs Supporting Risk-Based Performance Indicators

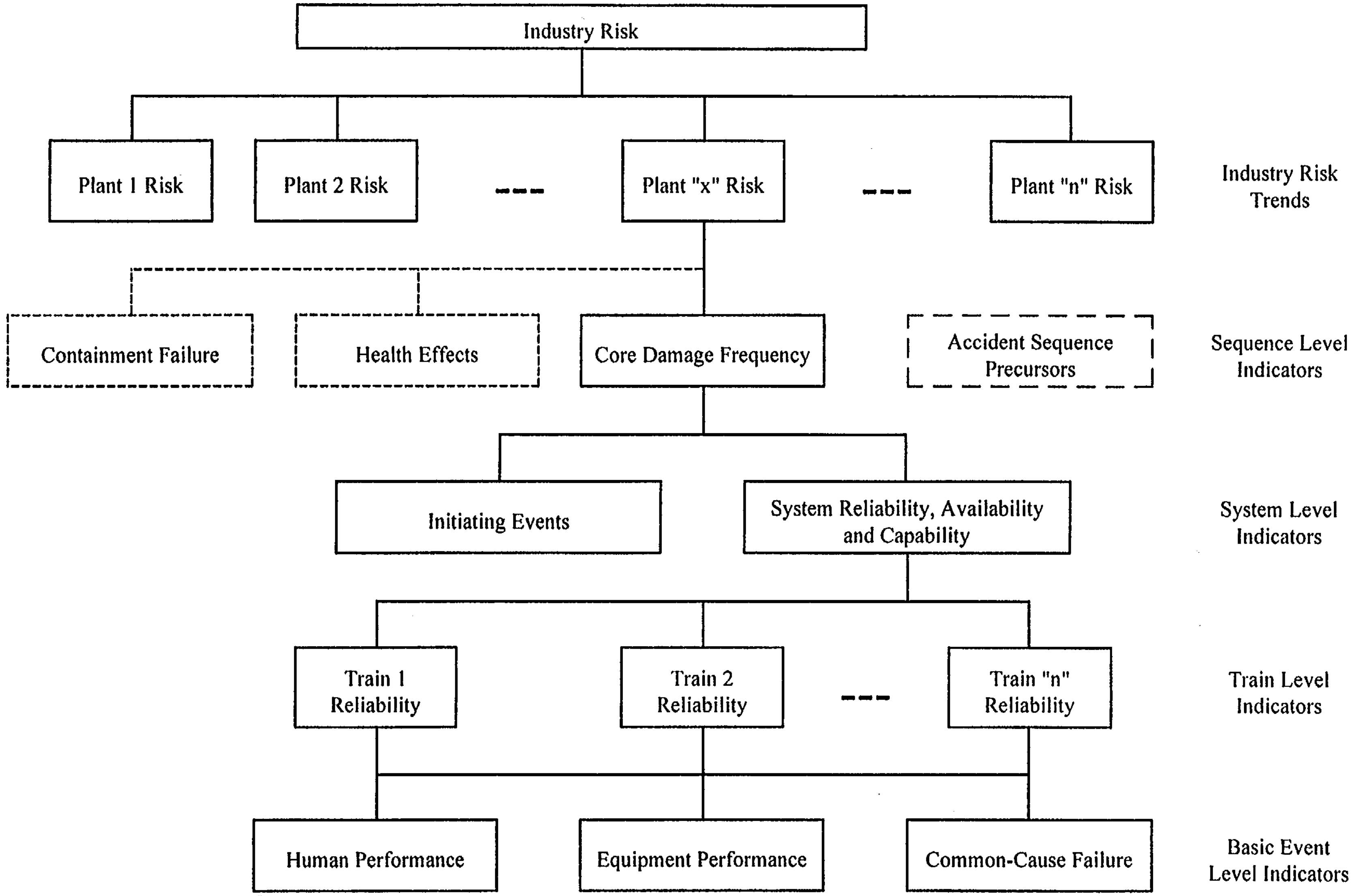

Figure 3 - Risk-Based Performance Indicators

Table 1 - Proposed Risk-Based Performance Indicators To Be Developed

Phase	Cornerstone	Operating Mode	Current Oversight Process Indicators	Proposed Risk-Base Performance Indicators
1	Initiating Events	Power	Unplanned Reactor Scrams Reactor scrams with loss of normal heat removal Unplanned reactor power changes	Fire frequency Loss of feedwater frequency Loss of ultimate heat sink frequency Loss of offsite power frequency
		Shutdown/ Refueling	Shutdown margin (future)	Fire frequency Loss of offsite power frequency Loss of residual heat removal system frequency Loss of inventory frequency
	Mitigating Systems	Power	Safety system unavailability Safety system functional failures Safety system unreliability (future)	Basic event level reliability and availability - Pumps (motor and turbine) [key risk systems] - Valves [key risk systems] - Common-cause failure - Operator performance in response to transients Train level reliability and availability - Emergency diesel generators - Auxiliary feedwater pump trains - Auxiliary feedwater injection paths - PWR High pressure injection pump trains - Component cooling water and service water pump trains System level reliability - On-site emergency ac power - Auxiliary feedwater - PWR High pressure injection - BWR High pressure coolant systems - Component cooling water and service water
		Shutdown/ Refueling	Mitigation system availability (future)	Train level reliability and availability - Emergency diesel generators - Reactor vessel inventory control (e.g., high and low pressure injection) - Residual heat removal - Component cooling water and service water
	Integrated	Power	None	Core Damage Frequency
		Shutdown	None	Core Damage Frequency
2	Barriers	Power	Reactor coolant system specific activity Reactor coolant system identified leak rate Containment leakage	Containment spray system trains Containment cooling system trains Containment isolation system trains
		Shutdown/ Refueling	Reactor coolant system specific activity Reactor coolant system identified leak rate Containment leakage	Containment spray system trains Containment isolation components (e.g., equipment & personnel hatches)
	Integrated	Power	None	Core Damage Frequency + Barrier Integrity
		Shutdown	None	Core Damage Frequency + Barrier Integrity

Risk Perspectives Regarding Low-Power and Shutdown Operations

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Abstract

During the last few years the Nuclear Regulatory Commission (NRC) has undertaken significant efforts for developing guidance on using risk information in its decision-making process (risk-informed regulation). With respect to low-power and shutdown (LPSD) operations past and recent nuclear power plant operational experience indicates that the risk associated with LPSD could be significant. Therefore, it is important that the NRC considers the risk from LPSD conditions in its risk-informed activities.

The NRC initiated in early 1999 a program with the objective to provide (or develop as necessary) an understanding of the risk-associated with LPSD operations sufficient to support risk-informed regulatory activities. The first phase of this program focused into gathering and evaluating information regarding LPSD risk. The results were documented in a "Perspectives Report" submitted to the Commission as attachment to SECY-00-0007. This paper summarizes the perspectives presented in the report addressing four topics: the significance of LPSD risk, the methods and tools currently available to evaluate and manage LPSD risk, the limitations of current methods and tools to support risk-informed regulation. The paper also includes recommendations for addressing the issues identified along with proposed milestones and schedule for the second phase of this program.

Introduction

Both nuclear power plant operational experience and probabilistic risk assessments (PRAs) indicate that the risk associated with LPSD operations could be significant. In the past ten years important events have occurred and a number of NRC, industry, and international studies have concluded that significant frequencies of reactor core damage could occur from accidents initiated during LPSD operations. This perspective has led to several initiatives to ensure safety during such operations, including:

- Reactor licensees have developed and applied methods for better managing plant safety during outages.
- The American Nuclear Society (ANS) has initiated work to develop standards for qualitative and quantitative methods for assessing LPSD risk.
- The NRC has issued generic letters and information notices and performed supporting risk studies with respect to LPSD conditions.

At the same time, significant efforts have been undertaken on using quantitative risk information in the NRC's decision-making processes (risk-informed regulation). The Commission published its PRA Policy Statement (Ref. 1) encouraging the use PRA and the expansion of "the scope of PRA applications in all nuclear regulatory matters to the extent supported by the state-of-the-art in terms of methods and data." The

NRC staff has published Regulatory Guide 1.174 (Ref. 2), which provides guidance for the use of risk information in regulatory activities and started using risk information in all different aspects of regulation, including inspection and enforcement, performance evaluation and monitoring, and risk-informing 10 Code of Federal Regulations (CFR) Part 50. Therefore, the importance for the NRC having the capability to incorporate the risk associated with LPSD operations into risk-informed regulatory activities has increased.

Although all nuclear power plants have performed a Level 1/2 PRA for the reactor at full power as part of the Individual Plant Examination Program (Generic Letter 88-20), few plants have performed a PRA for the reactor in low-power or shutdown conditions. The Advisory Committee on Reactor Safeguards (ACRS) (Ref. 3) expressed concerns that the NRC staff did not have an adequate understanding of the risk associated with LPSD conditions for rendering timely regulatory decisions of licensee submittals and, in follow-on discussions, recommended that the necessary benchmarking of risk during LPSD operations should be established. The staff, as part of the NRC budget for fiscal year 1999, recommended to the Commission (and received approval for) additional research studies on LPSD operations. As a result, this program was initiated in early 1999.

Objective

The objective of this program is to provide (or develop as necessary) an understanding of the risk associated with accidents during LPSD conditions sufficient to support risk-informed regulatory activities.

Approach

The first phase of this program included the following activities:
- Gathering information on domestic and international LPSD risk assessment methods and results,
- Meeting with stakeholders to discuss LPSD PRA results, current methods used by licensees to manage LPSD risk, and potential guidance, methods, and tool development needs,
- Evaluating strengths and limitations of available methods and tools for use in regulatory applications,
- Identifying and prioritizing tasks to address identified limitations, and
- Developing a multi-year plan for performing these second-phase tasks.

The results of the first phase were documented a "Perspectives Report," attached to SECY-00-0007 (Ref. 4) and are summarized in this paper. Perspectives were developed in the following areas:
- the significance of LPSD risk,
- current methods and tools for evaluating LPSD events,
- adequacy of current methods and tools to support risk-informed regulatory activities, and
- recommendations

Perspectives on the significance of LPSD risk

Perspectives on the significance of LPSD risk were obtained from a review of operational events and LPSD risk related studies (both domestic and international). The following are overall perspectives.

Numerous events that are potentially risk-significant have occurred at U.S. plants during LPSD conditions. These events have required mitigation with plant safety systems to prevent core damage. These events combined with the plant response provide relatively high estimated conditional core damage probabilities (1E-4 to 1E-3) and therefore, from a risk perspective, warrant consideration. In other words, initiating events unique to LPSD suggest the contribution to risk during LPSD from the design and operation of the plant is significant. Table 1 summarizes perspectives obtained from reviewing events occurred since 1987.

Table 1. Summary of perspectives on LPSD events.

- Similar events are occurring across the industry.
- The events have generally involved:
  - — loss of shutdown cooling,
  - — loss of coolant,
  - — loss of offsite power, and
  - — loss of power supplies other than those initiated by loss of offsite power.
- The specific causes of the events tend to be plant-specific.
- The majority of the events include human factors involving:
  - — personnel errors, and
  - — deficient procedures.
- Plant configuration and/or incomplete operator knowledge may contribute to an event and may limit the capability to mitigate an event.

There remains a broad consensus that the frequencies of core damage accidents initiated during LPSD operations can be significant. Domestic (NRC and industry) and international studies reviewed by the NRC staff, as well as comments provided by stakeholders in public meetings, consistently identify potential accidents initiated during some portions of LPSD operations as significant contributors to the total core damage frequency (CDF) from licensed nuclear power plants. Table 2 provides a summary of perspectives obtained from reviewing LPSD risk studies.

Table 2. Summary of Perspectives from LPSD Risk Studies

- Important initiating events include
  - — loss of shutdown cooling,
  - — loss of coolant,
  - — loss of offsite power, and
  - — internal fires
- CDF and risk estimates can be comparable to full power values.
- Human actions and the uncertainty associated with them are major contributors to LPSD risk.
- Dominant accident sequences usually involve plant conditions where:
  - — decay heat is still high,
  - — water level is reduced, and/or
  - — equipment is being maintained.
- The more important plant operational states (POSs) have the same characteristics as the dominant sequences.
- Conditional LPSD risk (per hour) is comparable (or higher) than conditional full power risk.
- Conditional (per hour) risk cannot be directly scaled to risk per year (risk based on being in a plant operational state (POS) for a full year).
- Physical separation of equipment is an important factor in determining whether fire or flood will be important to risk.
- Equipment design and plant location are important factors in determining the significance of seismic events.
- Transition risk (i.e., risk associated with shutting down a plant to accomplish some activity) associated with repair of failed equipment can exceed the risk associated with continuing to operate while repairing the equipment.

Perspectives on current methods and tools for evaluating LPSD risk

Perspectives were developed by surveying qualitative and quantitative approaches as well as tools currently available for evaluating LPSD risk. The following are overall perspectives.

Licensees have developed qualitative and quantitative methods and tools for managing safety during LPSD operations (Refs 5-8). To manage LPSD risk, industry guidance has been developed and implemented which provides a qualitative means for licensees to manage safety during outages. Also, over one-half of the licensees supplement this qualitative guidance with some type of quantitative probabilistic risk analysis tools and information. Table 3 provides a summary of perspectives on industry LPSD risk methods and tools surveyed by the NRC.

Table 3 Perspectives of LPSD Risk Methods

Most of the industry methods and tools were developed for configuration risk management (CRM) purposes.
- CRM is primarily performed using qualitative defense-in-depth approaches using NUMARC 91-06 (Nuclear Management and Resources Council) guidelines.
- PRA is used for CRM purposes to augment defense in depth by more than one half of the U.S. utilities.

- LPSD PRA risk models typically cover refueling outages only.
- The scope of an LPSD PRA typically is an accident analysis for internal events (excluding fires and floods) during cold shutdown and refueling operational states.
- Risk measure metrics include boiling frequency, CDF, and fuel damage frequency.
- Few LPSD PRAs include an accident progression or consequence analysis.
- In principle, the software tools available for LPSD risk analysis can be used to develop a model with any desired level of detail or breadth of scope.

Perspectives on the adequacy of current methods and tools to risk-informed regulation

The methods and tools currently available for evaluating LPSD risk were examined from the perspective of their adequacy to support risk-informed regulation. The following are overall perspectives.

Current LPSD PRA methods provide a strong foundation for considering LPSD accident risks in regulatory activities. However, they need to be supplemented to permit their use in regulatory applications such as risk-informed license/requirement changes and oversight. Table 4 summarizes the limitations of current LPSD PRA methods to support risk-informed regulatory activities.

Table 4. Summary of limitations of current LPSD PRA methods to support risk-informed regulatory activities

Initiating Event Analysis • Draindown event causes and frequencies are not well understood. • Limited assessment of internal flood/fire and external events.

Accident Sequence Analysis
• Streamlining of accident sequence analysis
• Several accident sequences are poorly understood
    • spent fuel pool misloading
    • cold over pressurization
    • crane failure during heavy lifts
    • fast-acting reactivity insertions

Success Criteria • Current application of full-power success criteria to LPSD conditions may yield overly simplistic and inaccurate insights regarding risk contributors. • Thermal-hydraulic methods developed for full-power PRA may be inefficient when applied to LPSD condition. • Many LPSD thermal-hydraulic analysis are based on simplistic calculations.
Systems Analysis • Criteria and guidance for modifying full-power system models to match LPSD conditions have not been standardized. • Many utilities express concern that a meaningful baseline LPSD configuration could be defined for risk-informed purposes.
Data Analysis (Component Failure Models and CCF) • Appropriateness of using standby failure data for extended operations is not understood. • LPSD configurations may be significantly different than the basis for which full-power models were developed.
Human Reliability Analysis (HRA) • Application of full-power recovery models may yield overly simplistic and inaccurate insights regarding risk contributors. • Errors of commission • Transition risk
Accident Sequence Quantification • Only point estimate quantification is performed.
Level 2/3 • No assessment of Level 2 or 3 risk is done. • Source term and release mechanisms relevant to LPSD may not be well understood.

There is a strong need to develop better guidance for licensees and NRC staff on how to use current LPSD risk analysis methods in risk-informed regulatory activities. The staff believes that many of the limitations of current methods could be overcome through the development of guidance. This guidance would address technical issues such as how current qualitative or quantitative methods should be adapted for LPSD risk analysis as well as regulatory process issues such as how such methods can be used in specific risk-informed regulatory activities. It is anticipated that the ANS work to develop LPSD risk standards will play a key role in the development of such guidance.

In selected areas there is also a need to improve methods and tools for assessing LPSD accident risk. The staff's evaluation of documented LPSD risk studies, and comments provided by stakeholders, reveal several areas in which methods and tools should be developed or improved for use in risk-informed regulatory decision making.

A discussion of the areas that need either guidance or method development is provided below.

Guidance or Method Development Needs

Improve the treatment of internal fire and flood and seismic initiators. Previous LPSD analyses have shown that risk from these initiators can be important contributors to specific plants (e.g., an analysis of Surry indicated that fire was a dominant contributor to CDF). As such, it is important to identify the shutdown-specific conditions and activities (i.e., issues) that affect internal fire flood and seismic analyses. Once done, issues that are already adequately examined using current fire, flood, and seismic methodologies should be identified. The effects on CDF and risk of the remaining issues should be prioritized. For those issues deemed a high priority, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions.

Develop techniques for performing a simplified Level 2 risk analysis. To help ensure a more complete understanding of the risk (i.e., beyond CDF) from LPSD conditions and to allow a more direct comparison of the LPSD risk (i.e., public risk) with that from full power, the salient features from the current full-power simplified Level 2 risk analysis process applicable to shutdown conditions should be identified. LPSD conditions should be examined to identify areas requiring additional research sufficient to support these salient features. The research necessary to develop a shutdown-specific simplified Level 2 risk analysis should be performed. The techniques to incorporate a simplified Level 2 risk analysis into a shutdown analysis should be developed or current techniques enhanced, as required.

Improve the treatment of unplanned outages. Generally, LPSD analyses have not examined the risk from unplanned outages. To enhance the completeness of information used in risk-informed decision-making, unplanned outages should be examined. An efficient method for assessing the risk associated with unplanned outages should be developed.

Improve the treatment of transition risk. To enhance the completeness of LPSD analyses, the degree to which transition risk is currently accounted for in shutdown analyses should be identified and the issues associated with transition risk should be investigated and prioritized. For those issues that are deemed a high priority, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions.

Improve the treatment of fast-acting reactivity insertions. At the Sizewell B facility, this issue was found to contribute about 20% of the total fuel damage frequency during shutdown. To help ensure the adequate treatment of this issue, the significance of the following set of issues should be investigated: (1) potential pathways for un-borated water injection, (2) mitigative effects of mixing in the core region and mixing in the piping, and (3) maximum damage that could be expected if a slug of un-borated water moves through the core region. For those issues that are determined to be important, develop or enhance current techniques to incorporate them into an analysis of shutdown conditions, including a quantification methodology.

Improve HRA used for LPSD conditions. Both the NRC and industry analyses have identified the importance of human actions. To help ensure that human actions are adequately analyzed during LPSD conditions, LPSD issues that can affect HRA should be identified and prioritized. For those issues deemed high priority, develop or enhance existing HRA techniques to incorporate into LPSD analyses. Ascertain whether errors-of-commission are important to LPSD risk. If important, develop or enhance current techniques to efficiently model and incorporate errors-of-commission into LPSD analyses.

Improve treatment of crane failures associated with heavy load lifts inside containment. Because accidents resulting from crane failures are not usually included in LPSD analyses, develop or enhance existing techniques to model crane failures during LPSD conditions.

Update crane failure frequencies for heavy load lifts inside containment. To support the improved treatment of crane failures associated with heavy load lifts inside containment, the drop frequencies of nuclear grade crane drops should be updated to account for the past 20 years of nuclear grade crane operating experience.

Establish a LPSD baseline model. To support the use of LPSD risk assessment information in risk-informed regulatory decision-making, a baseline model for LPSD conditions should be developed. The model should account for forced and unplanned outages–at a minimum, accounting for historical forced and unplanned outages.

Improve the current understanding of draindown events and update frequencies. Because past analyses have found that these types of events can be important, past draindown events should be examined to identify factors that influence their occurrence. Initiating event frequencies and uncertainties used in industry and NRC-sponsored LPSD risk analyses should be updated to include post-1990 data. The frequencies should be adjusted to account for the factors that influence their occurrence.

Develop failure data for extended operations. To increase the realism necessary for risk-informed regulatory decision-making, currently available data should be examined to ascertain whether or not it is sufficient to produce failure rate estimates for components that experience extended periods of operation during shutdown conditions. If sufficient information is available, appropriate failure rates should be developed for the components.

Examine current thermal-hydraulic tools to ensure their efficient operation. Thermal-hydraulic calculations play an important roll in determining both success criteria and the time available for operators to respond to events. To increase the usefulness of thermal-hydraulic calculations, developers of the tools used to perform these calculations should be questioned to ascertain whether the tools function efficiently for shutdown conditions. Selected calculations should be performed to verify efficient operation during selected shutdown conditions. If necessary, areas where efficiency is lacking should be addressed by performing code modifications.

Investigate potential for spent fuel pool fuel misloading. Since plants are increasing the storage capacity of their spent fuel pools beyond their original limits and fuel misloading is not usually analyzed as part of a shutdown analysis, a method to assess the risk resulting from these activities should be developed.

Develop guidance on establishing success criteria. Because establishing success criteria is vitally important to correct accident sequence development, LPSD conditions should be examined to identify those that would affect the determination of success criteria. Guidance should then be developed to facilitate the appropriate consideration of these LPSD conditions.

Establish minimum requirements for defining a plant operational state. Defining plant operational states is one task in a LPSD analysis. Currently, different approaches are used, resulting in some analyses having a small number (e.g., 10 to 20) and some having a large number (i.e., much more than 20). To ensure that important parameters are considered when defining plant operational states for LPSD analyses, a workshop should be conducted to develop the minimum set of requirements necessary for defining plant operational states.

Enhance the streamlining of accident sequence analysis. Because LPSD risk has been found to be as important as full-power risk, and because resources should be used effectively to identify the more risk-significant conditions, current information should be examined to identify sets of conditions that generally make a plant state more risk-significant, or conversely, make a plant state less risk-significant. Characteristics of initiating events that can make them important even in a less risk-significant state should also be identified. Once identified, guidelines should be developed to determine which conditions and initiating events should be analyzed. To further enhance the wise use of resources, guidance should be developed on how many plant systems (capable of mitigating an accident sequence) to include in the accident sequence development process.

Develop guidance on how to use full-power models in LPSD analyses. Because of the importance of systems analysis to probabilistic risk assessment (PRA) and because conversion of full-power models for use in LPSD analyses is an efficient use of resources, guidance for converting full-power system models into models for use during LPSD analyses should be developed.

Develop guidance on the correct application of full-power common-cause failure (CCF) models to LPSD conditions. Because correct use or implementation of any PRA model is important, guidance should be developed on what common-cause factors should be reviewed to account for LPSD conditions. Furthermore, specific guidance should be developed on how to adjust full-power CCF models to account for LPSD conditions.

Provide guidance on simplified thermal-hydraulic calculations. Because thermal-hydraulic calculations play an important roll in determining both success criteria and the time available for operators to respond to events, it should be determined whether simplified thermal-hydraulic calculations are sufficient. If simplified calculations are deemed appropriate, then the minimum set of thermal-hydraulic modeling requirements for these simplified calculations should be identified.

Develop guidance on incorporating uncertainty and sensitivity analysis techniques into LPSD analyses. To enhance the usefulness of risk information from LPSD analyses and to provide a more complete understanding of what can be important to risk, guidance on using full-power uncertainty and sensitivity techniques as part of LPSD analyses should be developed.

Develop guidance on cold over-pressurization analysis. To enhance the completeness of LPSD analyses, guidelines for incorporating adequate cold over-pressurization models should be developed.

Recommendations

To provide a sound technical basis for addressing LPSD risk in regulatory activities, these research issues can be resolved either through guidance development, method and tool development, or standard development. The priority of the specific issue, however, is dependent on its importance to whether development of a plant-specific LPSD PRA, or some lesser type of LPSD model/approach is needed for the various risk-informed regulatory activities. Therefore, determining which research issues are needed for which approach is the first step in prioritizing the research issues.

By examining the importance of the identified research issues with respect to LPSD information needed for risk-informed regulation, the following four major tasks were recommended:

(1) Support development of an ANS LPSD PRA standard. Provide technical expertise in the drafting and finalization of the LPSD standard so that the standard meets NRC needs and can be used to support risk-informed regulatory activities. This task would also involve providing support in the resolution of technical issues important to the development of plant specific PRAs, such as the following:
   - requirements for defining a POS.
   - success criteria,
   - uncertainty and sensitivity analysis,
   - cold over-pressurization,
   - use of full-power models,
   - use of full-power CCF models,
   - internal floods, and
   - internal fire and external seismic

(2) Develop improved guidance for considering LPSD risks. This guidance would be developed for risk-informed licensing decisions (by, for example, supplementing the acceptance guidelines in Regulatory Guide 1.174 to specifically address LPSD risk as previously proposed in SECY-97-287 and Standard Review Plan Chapter 19 to specifically address guidelines for review of LPSD risk analysis), and would be utilized in developing proposed revisions to 10 CFR 50.

(3) Develop improved methods and tools for assessing HRA and Level 2 risk. These areas merit different consideration under LPSD conditions than how they are treated in full-power operation. In addition, a better understanding of these areas is needed because of the large uncertainties associated with them and because of their potential to directly impact staff risk-informed regulatory activities.

(4) Evaluate areas identified by the ACRS and other stakeholders as potentially important to risk. Development of improved methods or tools for these areas would be to the extent necessary to address the specific technical issue. To the extent possible and appropriate, this work would be performed in cooperative programs with the nuclear industry and NRC's international PRA research partners.[1]

Issues would include such items as:
- LPSD baseline model,
- Unplanned outages,
- Transition risk,
- Extended operation failure data,
- Fast acting reactivity insertions,
- Understanding of draindown events,
- Thermal-hydraulic tools,
- Spent fuel pool fuel mis-loading,
- Heavy load lifts, and
- Crane failure frequency.

The above recommendations are planned to be performed over the next two years subject to Commission approval.

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[1]NRC's International Cooperative PRA Research (COOPRA) Program, initiated in 1997 to identify and perform cooperative research, has a working group on LPSD risk assessment. PRA experts from fifteen countries participate in this working group.

REFERENCES

1. U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Activities: Final Policy Statement," *Federal Register,* Vol. 60, p. 42622, August 16, 1995.

2. U.S. Nuclear Regulatory Commission, "An Approach for Using PRA Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.

3. Letter from R.L. Seale, Chairman, Advisory Committee on Reactor Safeguards, to Chairman Shirley Ann Jackson, USNRC, " Establishing a Benchmark on Risk During Low-Power And Shutdown Operations" April 18, 1997.

4. U.S. Nuclear Regulatory Commission, "Proposed Staff Plan for Low Power and Shutdown Risk Analysis Research to Support Risk-Informed Regulatory Decision Making," SECY-00-0007, January 12, 2000.

5. Nuclear Management and Resources Council, Inc., "Guidelines for industry Actions to Assess Shutdown Management," NUMARC 91-06, 1991.

6. Electric Power Research Institute, "ORAM-SETNINEL™ Software Version 3.3," SW-112894-CD M Palo Alto, California.

7. Electric Power Research Institute, "EOOS, A Tool for Risk Awareness," Version 3.0, November, 1999.

8. SCIENTECH, INC, "Safety Monitor™ Software Version 2.00 Verification and Validation (V&V)," RZ23AF.05c, Kent, Washington, September 1998.

Risk-Informing 10 CFR 50

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The United States Nuclear Regulatory Commission is improving its regulatory processes by making greater use of methods and results of probabilistic risk analysis. One important element of this improvement is the modification of reactor safety regulations to better balance the imposed regulatory burden of specific regulations or sets of regulations with their safety importance. Two staff projects to implement these modifications are described in this paper: one which will modify the scope of "special treatment" regulations (e.g., equipment qualification); and a second which will identify potential changes to technical requirements (e.g., on hydrogen control during accidents).

Introduction

In the past several years, the NRC staff's work to make its reactor regulatory requirements and processes more "risk-informed" has made several significant steps forward. This work has resulted in:

- Development and application of guidance (Ref. 1) on the use of risk information, in concert with traditional engineering information, for changes to the licensing bases of commercial nuclear power plants;

- Preliminary revision of the reactor oversight process (which includes inspection, enforcement, and assessment) to make greater use of objective performance indicators and to focus staff and licensee resources on the most safety-significant issues and plant activities (Ref. 2); and

- Initiation of a program to modify the NRC's fundamental reactor regulations, contained in 10CFR50, to provide:
  - risk-informed changes to "special treatment" requirements (Ref. 3), and
  - risk-informed modifications to technical regulations (Ref. 4).

This paper discusses the staff's work with respect to this last program.

Background

In December 1998, the staff proposed three options for modifying regulations in 10CFR50 to make them risk-informed and defined four associated policy issues for Commission consideration (Ref. 5). These options were:

1. Continue with ongoing rulemakings, but make no additional changes to Part 50,
2. Make changes to the overall scope of systems, structures, and components (SSCs) covered by those sections of Part 50 requiring special treatment (such as quality assurance, technical specifications, environmental qualification, and 50.59) by formulating new definitions of safety-related and important-to-safety SSCs, and
3. Make changes to specific requirements in the body of regulations, including general design criteria (GDCs).

In a June 1999 Staff Requirements Memorandum (SRM) (Ref. 6), the Commission: (1) approved proceeding with the current rulemakings in Option 1, (2) approved implementing Option 2, and (3) approved proceeding with a study of Option 3.

Rulemaking on Special Treatment Requirements

The staff's work to implement Option 2 has led to the development of a proposed rulemaking plan, provided to the Commission for review and approval in October 1999 (Ref. 3). The purpose of this rulemaking is to develop an alternative regulatory framework that enables licensees, using a risk-informed process for categorizing SSCs according to their safety significance (i.e., a decision that considers both traditional deterministic insights and risk insights), to reduce unnecessary regulatory burden for systems, structures, and components (SSCs) of low safety significance by removing these SSCs from the scope of special treatment requirements. In the process, both the NRC staff and industry should be able to better focus their resources on regulatory issues of greater safety significance. This framework should improve regulatory effectiveness and efficiency, and contribute to enhanced plant safety. To accomplish this goal, it is necessary to amend the governing regulations. The current regulations use terms such as "safety-related," "important to safety," and "basic component" to identify the groups of SSCs and associated activities that require "special treatment." This rulemaking will build into the regulations an alternative that offers licensees the flexibility of utilizing a risk-informed process to evaluate the need for special treatment. This risk-informed process will ensure that risk insights will be used in a manner that complements the NRC's traditional deterministic approach. The risk-informed approach will be consistent with the defense-in-depth philosophy, will maintain sufficient safety margins, will ensure that any increase in core damage frequency or risk is small and consistent with the safety goal policy statement, and will include a performance measurement strategy. The risk-informed framework will also be aligned to the NRC's new oversight process (Ref. 2) by incorporating the cornerstones from the reactor safety and radiation protection safety areas into the SSC categorization process.

A graphical depiction of the changes that are expected to result from a risk-informed re-categorization of SSCs is illustrated in Figure 1. The figure depicts the current safety-related versus nonsafety-related SSC categorization scheme with an overlay of the new risk-informed categorization. The risk-informed categorization would group SSCs into one of the four boxes in Figure 1.

Box 1 of Figure 1 contains safety-related SSCs that a risk-informed categorization process concludes are significant contributors to plant safety. These SSCs are termed risk-informed safety class 1 (RISC-1) SSCs. SSCs in this box would continue to be subject to the current special treatment requirements. In addition, it is possible that some of these SSCs may have additional requirements concerning reliability and availability, if attributes which cause an SSC to be safety significant are not sufficiently controlled by current special treatment requirements.

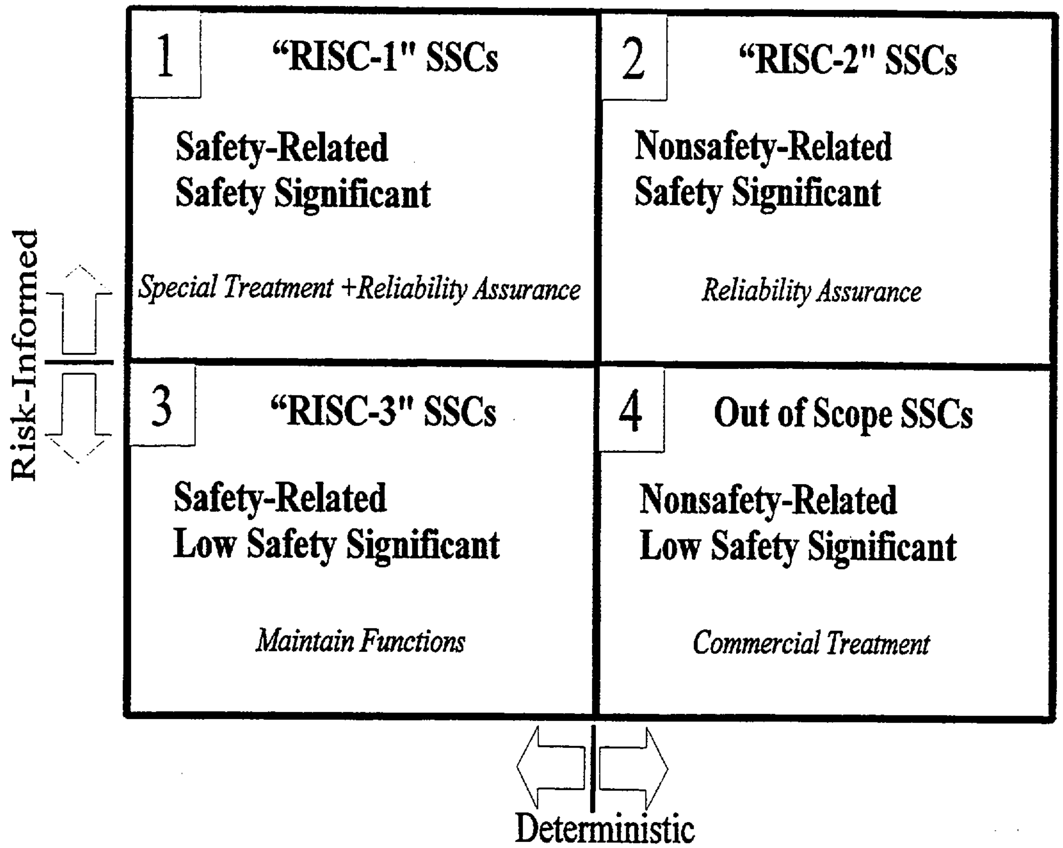

Figure 1: Diagram of Categorization and Treatment

Box 2 depicts the SSCs that are nonsafety-related, and that the risk-informed categorization concludes make a significant contribution to plant safety. These SSCs are termed RISC-2 SSCs. Examples of RISC-2 SSCs could include the station blackout emergency diesel, startup feedwater pumps, or SSCs that function for pressurized water reactor (PWR) "feed and bleed" capability. For RISC-2 SSCs, there will probably need to be requirements to maintain the reliability and availability of the SSCs consistent with the probabilistic risk assessment (PRA). As discussed below, it is currently envisioned that 10CFR50.69 (i.e., the new rule) would contain the regulatory treatment requirement for RISC-1 and RISC-2 SSCs regarding the reliability and availability of these SSCs.

Box 3 depicts the currently safety-related SSCs that a risk-informed categorization process determines are not significant contributors to plant safety. These SSCs are termed RISC-3 SSCs. The rulemaking would revise Part 50 to contain alternative requirements (per §50.69) such that RISC-3 SSCs would no longer be subject to the current special treatment requirements. For RISC-3 SSCs, it is not the intent of this rulemaking to allow such SSCs to be removed from the facility, or to have their functional capability lost. Instead, the RISC-3 SSCs will need to receive sufficient regulatory treatment such that these SSCs are still expected to meet functional requirements, albeit at a reduced level of assurance. The staff may determine that this level of assurance can be provided by licensee's commercial grade programs. As discussed below, it is currently envisioned that §50.69 would contain the regulatory treatment requirements for RISC-3 SSCs.

Box 4 depicts SSCs that are nonsafety-related and continue to be categorized as not being significant contributors to plant safety. These SSCs are out of scope of both current special treatment and any future regulatory controls of §50.69. The functional performance of these SSCs is controlled under the licensee's commercial grade program (no change from the current requirements).

As described in Reference 3, the staff is recommending a rulemaking approach that would include development of a new Part 50 rule. This rule would be supported with an appendix that utilizes new terminology as presented in Figure 1. The staff is recommending this approach in lieu of modifying the definition of "safety-related" and defining "important to safety" as was suggested in Reference 5 because if the current terminology is redefined to include a risk-informed part and voluntary and selective implementation is allowed, the meaning of "safety-related" and "important to safety" would be licensee and rule-specific. The staff believes that this outcome would result in confusion among both the staff and industry. With the use of new terminology, it would be immediately apparent whether a licensee was using the risk-informed alternative or the current requirement. The staff's proposed terminology, as previously described, is risk-informed safety class (RISC)1, 2, and 3.

The rulemaking approach includes two parts. The first part is a new rule, 10CFR50.69, that will allow the use of the new risk-informed categorization for the regulations identified within that rule. Section 50.69 will require that licensees use a method that complies with criteria in a new Appendix T "Categorization of SSCs Into Risk-Informed Safety Classes," or is otherwise found acceptable by the staff, to identify the appropriate SSCs for each risk-informed safety class. Section 50.69 will also provide requirements for regulatory treatment depending on the risk-informed safety class. Licensees would be allowed to use the risk-informed approach for any of the rules, or sets of rules as appropriate, that are identified in §50.69. The second part is a new 10CFR50 Appendix T that provides the criteria and categorization processes to properly identify safety significant SSCs that require special treatment. An objective of this rulemaking is to attempt to establish criteria in Appendix T such that licensee's who satisfy those requirements will be able to implement the risk-informed alternative with little or no prior staff review of the categorization process.

Reference 3 includes a proposed schedule for developing a proposed and final rule, including the following:

- A proposed rulemaking package submitted to the Commission for approval in September 2000.
- A final rulemaking package submitted to the Commission for approval in October 2001.
- Licensee implementation could begin in March 2002.

Study of Changes to Technical Requirements

The staff's work with respect to Option 3 has led to the development of a study plan, provided to the Commission for review and approval in November 1999 (Ref. 4). As discussed in that paper, the staff's work will consist of two phases: an initial study phase (Phase 1) in which recommendations to the Commission on proposed changes will be made; and an implementation phase (Phase 2), where changes recommended in Phase 1, and approved by the Commission, will be made.

In Phase 1, the staff will study the ensemble of technical requirements contained in 10CFR50 (and its associated implementing documents such as regulatory guides and standard review plan sections) to (1) identify individual or sets of requirements potentially meriting change; (2) prioritize which of these requirements (or sets of requirements) should be changed; and (3) develop the technical bases to an extent that is sufficient to demonstrate the feasibility of changing requirements. This work will result in a set of recommended changes to the requirements and associated priorities for implementation. These recommendations will be provided to the Commission for review and approval.

The staff's Phase 1 work will be conducted in the following manner:

- The staff's work will focus on providing a better balance to the Part 50 technical requirements among needed defense-in-depth and safety margins as well as risk. This improved balance will be achieved by systematic consideration of the Part 50 requirements. The staff's approach to risk-informing Part 50 will necessitate a broad assessment of Part 50 requirements, rather than a review of individual regulations. As such, the staff's work may involve changing regulations in sets, rather than individually. That is, risk-informing Part 50 may involve relaxing requirements in some areas in combination with increasing requirements in other areas.

- The set of safety principles established in Regulatory Guide (RG) 1.174 (Ref. 1) will be applied to possible changes to requirements studied in this phase. That is, the staff expects that the changes to requirements resulting from this work would be consistent with the defense-in-depth philosophy, would maintain sufficient safety margins, would be performance based to the extent possible, and would result in safety improvements or only small increases in risk, and would reduce any unnecessary burden. This approach would also ensure that adequate protection continues to be maintained.

- The study will focus on potential changes to the technical requirements associated with 10CFR50. Since the basis for these requirements may be contained in the regulations themselves or in supporting regulatory guides, Standard Review Plan sections, branch technical positions, or other documents, all such documents should be reviewed and, as necessary, be considered for change.

- The study may lead to recommendations which either reduce existing requirements (by modification or elimination) or impose new requirements.

- The criteria applied in this study for risk categorization will build upon and be consistent with those being used in Reference 3. This work will also build upon and be coordinated with the risk-informed plant oversight program (Ref. 2).

- The criteria established in this study with respect to needed quality of a licensee PRA will be consistent with those proposed in Reference 3 and RG 1.174. PRA standards, either developed by standards-setting organizations (e.g., ASME and ANS) and endorsed by NRC, or developed by NRC, are intended to be important mechanisms for ensuring needed quality.

- The fundamental concept of using a set of design basis accidents (DBAs), and categories of anticipated operational occurrences (AOOs), to set the design of licensed reactors is expected to be retained. However specific DBAs or AOOs may be modified or eliminated or new DBAs or AOOs established.

- The principal focus of this work is on the current set of licensed reactors. As such, the staff intends to study changes to specific existing requirements, rather than making a *de novo* rewrite of Part 50.

However, one factor in the staff's prioritization process will be the potential impact on future reactors, so that potential regulatory changes that impact both current and future plants will receive higher priority than those only affecting current reactors. Those changes affecting only future plants will be of lowest priority.

Using this approach, the staff will perform Phase 1 in three tasks. These three tasks include:

<u>Task 1: Identification of Candidate Changes to Requirements and Design Basis Accidents</u> This task will provide a first screening of the technical requirements in 10CFR50, implementing documents, and DBAs. This screening will use three criteria to identify the best candidates for change. These three criteria are: frequencies of the initiating event and event scenarios; risk contributions of the scenarios and systems, structures, and components (SSCs); and extent of excessive conservatism or non-conservatism in associated methods, assumptions, or acceptance criteria. Each of these criteria will be used to identify requirements and DBAs which appear to have a frequency, risk, or conservatism which is either inordinately high or low.

<u>Task 2: Prioritization of Candidate Changes to Requirements and Design Basis Accidents</u> This task will provide a prioritization of candidate changes identified in Task 1. Prioritization criteria to be used include rough estimates of the values and impacts of the candidate change (including values in safety benefit and burden reduction, and impacts in costs to the NRC and the licensee to make the change); and the practicality of the candidate change. Evaluation of the benefits will include consideration of the population of plants expected to be impacted by the change (including both currently licensed reactors and future plants). Backfitting issues associated with candidate changes will also be evaluated at this time.

<u>Task 3: Identification of Recommended Changes to Requirements</u> This task will establish the scope and feasibility of implementing candidate changes identified in Task 1 and prioritized as high in Task 2. Results of this evaluation will be provided to the Commission in the form of recommendations on specific changes to 10CFR50 and its associated implementing documents.

The staff intends to test this process using at least two example Part 50 modifications, one involving the modification of a single requirement (e.g., hydrogen control requirements in 10CFR50.44) and one involving modification of a set of related requirements (e.g., requirements related to special treatment of SSCs). After completion of these test applications, and stakeholder comment, the staff will update the plan and begin a more extensive review of Part 50 requirements.

The staff expects to complete these test applications in early 2000. Recommendations on specific modifications to Part 50 (i.e., completion of Phase 1, Task 3) will be provided to the Commission in December 2000.

References

1. USNRC, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Current Licensing Basis," Regulatory Guide 1.174, July 1998.

2. USNRC, "Recommendations for Reactor Oversight Process Improvements," SECY-99-007, January 8, 1999.

3. USNRC, "Rulemaking Plan for Risk-Informing Special Treatment Requirements," SECY-99-256, October 29, 1999.

4. USNRC, "Proposed Staff Plan for Risk-Informing Technical Requirements in 10 CFR Part 50," SECY-99-264, November 8, 1999.

5. USNRC, "Options for Risk-Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities," SECY-98-300, December 23, 1998.

6. Memorandum from Annette Vietti-Cook, USNRC, to William D. Travers, USNRC, "Staff Requirements - SECY-98-300 - Options for Risk-Informed Revisions to 10 CFR Part 50 - Domestic Licensing of Production and Utilization Facilities," June 8, 1999.

PRESSURIZED THERMAL SHOCK -- A PROGRAM TO REVISIT THE TECHNICAL BASIS

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In the late 1970's, a challenge to the integrity of embrittled reactor pressure vessels (RPV) was identified that involved a rapid cooldown of the pressure vessel wall accompanied by either sustained high system pressure or a subsequent repressurization of the system. This challenge was termed pressurized thermal shock (PTS). Working from the state of the art at that time for probabilistic risk assessment, thermal hydraulic analysis, vessel fracture analysis, and material embrittlement estimation methods, the NRC staff developed one of the first performance-based, risk-informed regulations to assure the safe operation of the pressure vessel. The pressurized thermal shock rule, 10 CFR 50.61, established an embrittlement screening criterion above which licensees would be required to demonstrate that their pressure vessels could be operated safely. The screening criterion was derived using both conservative deterministic analyses and risk concepts that established an acceptable probability of vessel failure that implicitly considered the conservative nature of the deterministic analyses **[1-4]**. The NRC subsequently developed Regulatory Guide 1.154 **[5]**, on the format and content of analyses that could be used to demonstrate the continued safe operation of pressure vessels that would exceed the PTS screening criterion.

In late 1989 and early 1990, the staff and the licensee, for the now decommissioned Yankee Rowe pressurized water reactor (PWR) plant, entered into an intensive evaluation of the pressure vessel for the Yankee Rowe plant **[6]**. The staff had identified a high level of embrittlement for the pressure vessel and both the licensee and staff turned to Regulatory Guide 1.154 as a basis for evaluating the integrity of the pressure vessel. During the course of that evaluation, the staff and industry identified a number of shortcomings and limitations in the regulatory guide methodology; chief among them being the technical basis for the fabrication flaw distributions used in the probabilistic fracture mechanics analyses. The Yankee Rowe evaluation, as well as the earlier evaluations that had formed the basis for the rule and regulatory guide, demonstrated that the way flaws were modeled, using 1970's non-destructive examination (NDE) data and resulting Marshall flaw distribution **[7]**, in the analysis dominated the uncertainty in the calculated probability of vessel failure. Other variables were also shown to be important, such as the embrittlement estimation methods, the fracture toughness curves, and the thermal hydraulics calculations.

In the intervening years, the staff and the industry have worked both separately and jointly to improve the data bases for the flaw-related variables and for the other key variables. Recent analyses that combined these advances to evaluate the potential impact on acceptable levels of pressure vessel embrittlement have shown that the significant levels of conservatism in the screening criterion can be reduced.

The NRC has initiated a program to revisit the technical bases for the PTS rule and to, potentially, propose a revision to that rule that would significantly reduce the unnecessary level of conservatism in the rule. This effort has been undertaken as a full-participatory activity where active participation by the public and industry in evaluating the changes in the relevant technologies has been sought.

Since the completion of the earlier study, new information has resulted in an improved analytical capability to evaluate PTS events. This includes improved embrittlement correlations, greatly improved knowledge to estimate original flaw density, size, orientation, and distribution, refinement of the probabilistic fracture mechanics code, improved understanding of criteria for flow interruption and flow stagnation, and fluid mixing behavior in the downcomer. This paper will describe the issues and the progress made in the areas of probabilistic fracture analyses, thermal hydraulics, and probabilistic risk analyses, and the plans for future work.

The Integrated Pressurized Thermal Shock (IPTS) studies were a series of studies performed in the early-mid 1980's as part of an NRC-organized comprehensive research project to confirm the technical bases for the pressurized thermal shock (PTS) rule, and to aid in the development of guidance for licensee plant-specific analyses. The research project consisted of PTS analyses for three PWR plants: Calvert Cliffs Unit 1 **[2]**, designed by Combustion Engineering; H.B. Robinson Unit 2 **[3]**, designed by Westinghouse; Oconee Unit 1 **[4]**, designed by Babcock and Wilcox.

The primary objectives of the IPTS studies were (1) to provide for each of the three plants an estimate of the probability of a crack propagating through the wall of a reactor pressure vessel (RPV) due to PTS; (2) to determine the dominant overcooling sequences, plant features, and operator actions and the uncertainty in the plant risk due to PTS; and (3) to evaluate the effectiveness of potential corrective actions.

I. <u>Probabilistic Risk Assessment Issues</u>:

Occurrence of a significant PTS challenge to the RPV requires all five of the following conditions: 1) rapid cooling of the primary system; 2) continuation of that cooling to a low temperature; 3) maintenance of high primary system pressure, or repressurization; 4) presence of a crack on or near the vessel's inner surface; and 5) significant irradiation embrittlement of the vessel material at the crack's location, which makes the material fracture toughness against brittle fracture to decrease substantially.

In order to cause the first three of these conditions, PTS events typically involve at least two (and quite often more than two) of the following abnormalities: **a)** overfeed of one or more steam generators; **b)** colder than normal feedwater to one or more steam generators (normal and/or auxiliary); **c)** higher than normal steam flow from one or more steam generators; **d)** excessive primary system pressurization by primary system charging pumps and/or safety injection pumps; **e)** colder than normal primary system injection flow; **f)** a small break in the primary system (SBLOCA) of such a size that significant primary system pressure can be maintained by the charging and/or safety injection pumps.

1. <u>Identification of Plants From Which PRA Results are Needed</u>.

The set of plants analyzed for risk resulting from PTS challenges should cover the range that presently exists of the various designs of the following systems, and of the variations that presently exist among their various operational procedures and training programs.

A. Systems that Significantly Affect PTS Risk

All systems, including their control and indication subsystems, that can contribute to the occurrence of any of the above abnormalities should be included in PTS risk evaluations. Those systems include: a) the normal and auxiliary feedwater systems; b) feedwater heaters; c) the steamline isolation system; d) the turbine flow control system; e) the steam pressure control and relief systems; f) primary system charging systems; safety injection systems; and g) primary system relief valves, letdown valves, and RCP seals (i.e., anything that can cause a SBLOCA).

The relative importance of these systems, and why that relative importance is different among the three plants that were analyzed in support of the original PTS rule, is discussed in some detail in the Oak Ridge National Laboratory (ORNL) Technical Letter Report **[8]**.

Dozens of such interactions, compensating effects, etc., are discussed in the ORNL letter report **[8]**. The type of systems understanding discussed in the letter report (i.e., of the reasons why certain systems are important on some plants but not so much on others), is necessary (ideally, that understanding should be extended to include all U.S. NPP designs, not just the original three plants that were analyzed for their PTS risk) to enable competent selection of a small set of plants whose PTS risk will cover the range present within the entire family of U. S. NPPs.

B. Operational Features that Significantly Affect PTS Risk

If a nuclear power plant is operated according to its procedures (normal, abnormal, and emergency), there is essentially no chance that a PTS-related challenge to vessel integrity will occur. It requires a combination of equipment failure and departure from those procedures (i.e., a human error or errors) to place the plant in a condition where the first three of the above five conditions (Section I) exist simultaneously.

Therefore, the plant's procedures, the plant personnels' training and experience, and the overall plant "safety culture" should be considered as an integral part of the PTS risk evaluation at any plant.

The effects of operator interventions, or the lack thereof, are also discussed in the above referenced ORNL letter report (thus, the comments made above regarding that report are also applicable to the subject of operational features).

It is anticipated that the set of IPTS plants previously evaluated for PTS risk **[2-4]**, along with the "generic" analyses that were performed to provide the initial basis of the PTS rule (Attachment A to SECY 82-465 **[1]**), and another plant (Palisades) will be found to adequately represent the scope of the above design and operational features of U.S. PWR plants. If not, hypothetical changes could be assumed for one of those plants in its design or operation, and that "hypothetical" plant could then be re-analyzed to more adequately cover any "holes" found in the set of analyses. In any case, it is necessary to conclude one of the following: that the present plants adequately cover the spectrum; or that certain hypothetical changes need to be assumed during a re-analysis (e.g., the HBR "hypo" plant previously re-analysed with a more embrittled vessel **[3]**).

2. Identification of Transients That Contribute Significantly to PTS risk.

Each plant analysis should identify all sequences that significantly contribute to PTS risk. The sequence and timing of all equipment failures and personnel errors that cause (i.e., initiate or exacerbate) each event should be specified. This will largely be provided by the four previous PTS PRAs, but there may be certain areas that should be revisited in a carefully selected manner.

For example, proper application of the ATHEANA human factors methodology could contribute to a better understanding of conditions that could significantly affect (i.e., initiate, exacerbate, or mitigate) the chance of a PTS event given those conditions (i.e., those "shaping factors"). Many such "shaping factors" may be hidden in the "balancing act" between the conflicting requirements of cooling the core and not overcooling the vessel. Proper personnel training and experience is vital to enable both goals to be met with an extremely low failure probability. It can make a significant difference whether the most recent industry (or plant specific) precursor event, and the retraining that often follows such events, was an overcooling or an undercooling precursor. The most recent event will be the one most likely avoided, at the possible expense of making the opposite event more likely to occur.

Application of ATHEANA to a limited number of carefully selected situations should be considered as part of this PTS rule re-evaluation. It is anticipated that any ATHEANA analyses will be conducted on a separate path (not part of the critical path for the PTS rule re-evaluation), and any applicable results of those analyses will be incorporated (i.e., included in the total PTS-related risk for the applicable plant or plants) as they become available.

3. Identification of Significant Design or Operational Changes Made to Plants Previously Analyzed for PTS Risk.

Design and operational changes are being considered by the NRC staff with technical support from a contractor (INEEL), and separately by a nuclear industry group which includes a representative from each of the four previously PTS-analyzed plants. These separate considerations will be integrated as discussed in Section I.3.C.

A. Recommendations will be developed, under INEEL's contract, concerning the manner and extent to which the earlier PTS/PRA analyses should be modified to support the revised PTS rule.

B. Preliminary discussions with the industry representatives from the four PTS analyses plants are being held to determine if system level changes have been made in these plants to make their final PTS risk more restrictive.

C. The findings in Sections I.3.A and I.3.B will be integrated and recommendations will be made regarding the transients to be analyzed/re-analyzed for each of the plants under consideration to support the development of revised PTS screening criteria. These recommendations will be based upon consideration of : 1) changes to the plants and the plant risk models; 2) review of operational data; 3) changes to PRA methods, tools, and practices since completion of earlier studies.

4. Conduct of Revised PTS-PRA Calculations (if necessary)

A. For the plants selected in Section I.1, with the transients specified in Section I.2 (including any additional transients or modifications to previous transients recommended), and including any input from significant changes identified in Section I.3, the revised PRAs will be conducted.

B. The result of each PRA will be specification of the sequence and (in a broad sense) the timing of all equipment failures and personnel actions that affect each transient that should be considered by this effort to revise the PTS screening limit.

5. <u>Conduct of TH Analysis to Determine Input to PFM Analysis</u>.

A. The above PRA results will be "binned" into groups that are expected to have similar thermal hydraulic characteristics (i.e., pressure/temperature histories). This may require some form of preliminary TH calculations in order to facilitate the TH binning (i.e., from the Section I.4.B-specified sequences of events, the TH characteristics may not be sufficiently obvious to facilitate efficient determination of the more detailed TH calculation that is needed for each of the several bins).

The total number of bins (i.e., the total number of detailed TH calculations) will be determined from optimization of calculation costs and the amount of conservatism that can be tolerated in the final revised PTS screening criterion. The larger the number of bins, the greater the cost, but the closer the "fit" between individual transients and their bin's conservatively representative TH calculation and the smaller the unnecessary conservatism.

B. The above specification of transients; their preliminary calculation and binning; the final, detailed TH calculation for each bin; the uncertainty in the TH results; and the final content of each bin (which transients are finally included within each bin) will be the subject of iterations between the PRA and the TH analysis groups.

II. <u>Thermal Hydraulics Issues</u>:

A comprehensive search for scenarios which are both probabilistically credible and physically significant is necessary. This requires input from PRA and feedback from thermal hydraulics and fracture mechanics to determine not only the probability of occurrence of a given sequence but also its risk significance. Scenario screening must integrate the knowledge from the three disciplines of PRA, thermal hydraulics, and fracture mechanics.

The task of the thermal hydraulics work is to provide the downcomer boundary conditions for the fracture mechanics analysis. The boundary conditions of interest are time-dependent system pressure, fluid temperature in the downcomer, and the convective heat transfer coefficient, h, from the fluid to the wall. Estimates of the uncertainties in these values are also required. It has been shown, however, that the uncertainty in the convective heat transfer coefficient is of lesser interest since the heat transfer process is conduction limited **[9]**.

System pressure is a global parameter that is more or less based on the equation for conservation of energy. In comparisons between code and experiment, the codes normally predict this parameter reasonably well, providing the break flow calculation is reasonably accurate. The fluid temperature in the downcomer is somewhat more difficult to predict. The 1-dimensional formulation for pipe flow means that the code calculates perfect mixing in pipes. For the downcomer, 2-dimensional behavior can be approximated, but care must be taken with RELAP to avoid excessive numerically predicted (but not physically realized) mixing using parallel channels with cross flow junctions.

The IPTS study and subsequent experiments and analysis have indicated that PTS scenarios can be categorized as follows:

A. Transients in which loop flow is sustained. In such cases no thermal stratification occurs in the cold leg, the downcomer is well mixed, and the downcomer does not cool substantially from its normal temperature of approximately 540 °F. Such transients include small break LOCAs where the break is sufficiently small that it along with HPI are insufficient to remove decay heat. For such events, the primary side pressure remains slightly above the secondary side pressure.

B. Transients in which loop flow stagnates. For these events the combination of break and HPI flow are sufficient to remove decay heat. The primary system temperature drops below the secondary side temperature and loop flow ceases. Cold leg stratification occurs and the downcomer cools according to relatively simple conservation of mass/energy relations where the flow into the downcomer consists of HPI flow and the flow out of the downcomer is related to the break flow.

C. Flow stagnation is possible in ways not envisaged in the IPTS study. First, all loops do not stagnate at the same time. Rather, they stagnate sequentially based on small differences in steam generator secondary conditions, geometry, or other factors. Secondly, some plants are of "2 x 4" configuration. Typically, one of the two cold leg return lines from a given steam generator will stagnate before the other.

D. Transients in which loop flow stagnates, and the break is sufficiently large for the system to depressurize below the accumulator pressure. This pressure is 600 psi for Westinghouse and Babcock & Wilcox plants, and 200 psi for Combustion Engineering plants. Because of the lower pressure of the Combustion Engineering plants, this event category is not expected to be applicable. For the other plants, however, the accumulators have the potential to inject much higher flow rates and cool the downcomer substantially more. Therefore, even though the system pressure is lower, this category of events cannot be disregarded for fracture analysis.

There are certain inherent limitations in the codes used to analyze the PTS events. The one-dimensional formulation of flow in pipes precludes any treatment of two fluid temperatures, such as thermal stratification. This led to calculation of unphysical flow oscillations in the original IPTS study. Secondly, the plume behavior in the downcomer and the momentum solution is generally of limited accuracy.

1. Current and Planned TH Activities

The TH plan of work is as follows:

A. Foremost, to ensure that for the risk significant events, the thermal hydraulic inputs are still operative. These are: system pressure, downcomer fluid temperature, and convective heat transfer coefficient, all as a function of time and with uncertainty estimates. It was noted earlier that the fracture mechanics calculations are not particularly sensitive to the value of the convective heat transfer coefficient. System pressure is normally calculated reasonably well. A mixed mean downcomer temperature can be calculated with good accuracy. For each plant, about 4 calculations are envisaged using TRAC or RELAP.

B. To support this effort, the former IPTS results will be reviewed in detail to determine their continued applicability. The cases that need to be updated will be determined (see detailed discussion in Section I). The phenomenon identification and ranking table (PIRT) process **[10]** will be utilized to define sensitivity studies to be performed to estimate uncertainty bounds.

C. Industry input will be obtained to ensure that the basis for the IPTS analysis in terms of plant design, control systems, and operator procedures is correct.

D. Information on loop flow stagnation will be updated to reflect current understanding. The Oregon State University facility will be used to generate PTS-specific data.

III. <u>Probabilistic Fracture Mechanics (PFM) Issues</u>:

In the years since the results of IPTS studies were published, the fracture mechanics models, the embrittlement database and embrittlement correlation, inputs for flaw distributions, and the probabilistic fracture mechanics (PFM) computer code have been refined. A major objective of present technical bases development activities is to determine the synergistic impact of these fracture-technology refinements together with updated PRA and TH systems analysis results on the probabilities of through–wall cracking failure of RPVs calculated for the earlier PTS studies.

1. Flaw Size and Flaw Density Distributions

It has been previously recognized that the flaw-related data have the greatest level of uncertainty of the input data for the IPTS plant studies **[2-4]** required to perform a PFM analysis of a RPV subjected to transient loading conditions such as PTS. In addition, in all previous analyses, all flaws were conservatively assumed to be located at the inner-surface of the vessel. To reduce the flaw related uncertainties and conservatism, the NRC has supported research to establish a better technical basis for estimating the flaw related data in the RPV material. The objective has been to determine the number, location, and sizes of flaws in the vessel material which are important input data into PFM analyses. One such research effort made use of an actual unused PWR vessel, named Pressure Vessel Research User Facility (PVRUF), which was located at the Oak Ridge National Laboratory. The Pacific Northwest National Laboratory (PNL), under contract to the NRC, has performed extensive nondestructive examination (NDE) and selective destructive examination (DE) of weld and adjacent base-plate materials taken from the PVRUF vessel **[11]**. While there was a considerably higher number of flaws found in PVRUF than had been previously postulated in the IPTS analyses, all flaws were sub-surface (embedded), i.e., no surface-breaking flaws were found.

The flaw size and density distributions **[11]** for PVRUF weld material is planned to be further supplemented with NDE/DE data from other RPV weld materials from Shoreham (boiling water reactor) and Midland (PWR) vessels. An expert-elicitation process is being planned to be performed to determine generic statistical distributions on flaw sizes and flaw locations in welds and base-plate materials. These generic flaw distributions and their uncertainty bands will be used in PFM analyses.

2. Revised Irradiation Embrittlement Correlations

The embrittlement correlations in Regulatory Guide 1.99, Revision 2 **[12]**, that are used in predicting the increased embrittlement over the life of the vessel (irradiation induced shift in RT_{NDT}), were based on analysis of Charpy data available in 1984. Since then, a large body of additional Charpy surveillance data have become available, and the understanding of embrittlement mechanisms has advanced. The resulting improved embrittlement correlations have recently been published **[13]**. Further refinement in the embrittlement correlation are being developed to include more recent embrittlement data, effect of long irradiation exposure time at vessel normal operating temperature, and statistical uncertainties in the predicted shift in RT_{NDT}.

3. Improvements in Fracture Mechanics Methodology

A number of significant improvements have been made, or being presently being carried out, in fracture mechanics analysis procedure that can be used efficiently in repetitive deterministic computation in which input variables are selected from a probabilistic distributions. Notable among them are:

A. Inclusion of available additional cleavage crack initiation toughness, K_{1c}, and crack-arrest toughness, K_{1a}, data that satisfy linear-elastic fracture mechanics validity criteria per ASTM standards, beyond what has been available in the ASME, Section XI, K_{1c} and K_{1a}, curves to develop statistically more rigorous and physically-based toughness curves and associated uncertainties for the RPV materials.

B. Reviewing plant-specific weld and base-plate material chemistry (nickel and copper weight percents) and the initial RT_{NDT} data for each plant considered in the PTS studies and determining appropriate statistical distributions for use in PFM analyses.

C. Stress intensity factor, K, solutions for semi-elliptical surface flaws of aspect ratios (≡ flaw depth/length) 2, 6, 10 and infinity that have been determined for the cladded-RPV using finite element computations in which the applied thermal and pressure induced stresses are decomposed into third-order polynomials.

D. The effect of clad to base-metal differential thermal expansion induced residual stress is determined from a more realistic, experimentally measured data.

E. Stress intensity factor, K, solutions for fully-elliptical sub-surface (embedded) flaws, (with aspect ratios of 2, 6 and 10) determined using an ASME Section XI methodology which has been validated selectively by finite element computations.

F. Inclusion of weld residual stresses distribution through-thickness in an RPV, that are determined from measurements, for use as an applied loading.

4. Beltline Vessel Fluence Calculations

An accurate calculation of fluence values in the RPV beltline region is very important for determining the effect of irradiation embrittlement on fracture toughness of the vessel materials. As such, fluence calculations for each of the plants is based on up-to-date information of the plant's cycle by cycle fuel loading history. Also, by using recent fluence calculation methods, the uncertainties in the determined best-estimate fluence values at the vessel inner-surface can be minimized.

5. Recent PFM Analysis Results

Application of PFM analysis advancements was made recently to an IPTS plant study **[14]** to determine the effect on the frequency of vessel failure (≡ probability of through-wall cracking failure per reactor year of operation). It was found that there is an order of magnitude reduction in the frequency of vessel failure when improvements, such as the flaws are distributed through the vessel thickness (rather than being conservatively assumed to be located at the inner-surface of the vessel), as per flaw distribution found in PVRUF vessel welds, revised embrittlement correlations, and the PFM analysis methodology are used.

IV. Quantitative Risk Acceptance Criterion

Recommendations regarding appropriate quantitative PTS risk acceptance criterion will be re-examined that are based on the frequency of through-wall cracking (≡ probability of through-wall cracking failure per reactor years of operation), reactor core damage frequency (CDF), and large early release frequency (LERF) criteria in light of more recent policies on PRA acceptance guidance such as the Safety Goal Policy Statement and RG 1.174 **[15]**.

V. PTS Re-analysis Plan

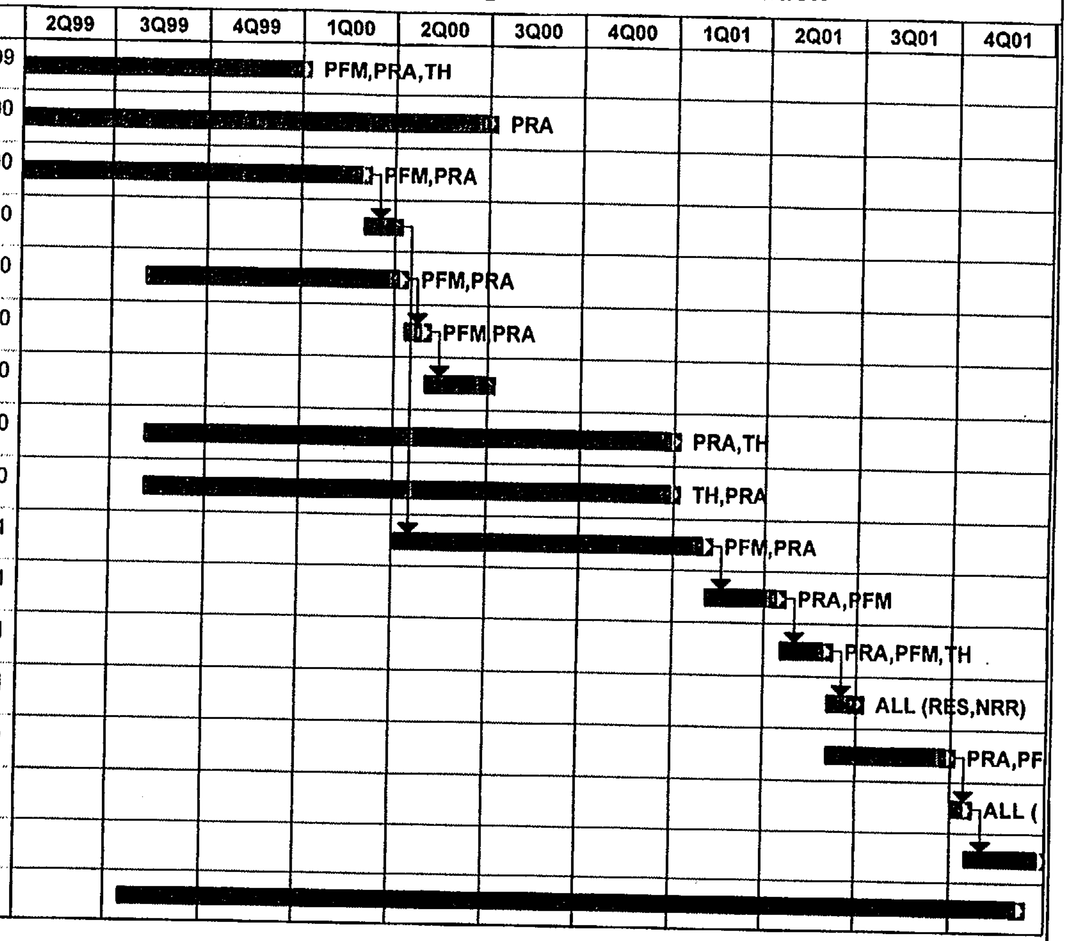
A comprehensive plan is being developed to provide a technical bases to re-evaluate the PTS screening criteria and 10 CFR50.61 PTS rule, which involves interdependent activities in PRA, TH and PFM areas. **Figure 1** shows significant milestones and estimated schedule for completing the technical bases development in the next 2 years. Involvement from the public and the industry is strongly sought at each step of the technical activities. Presentations to ACRS will be made at regular intervals to seek and implement their recommendations during the entire technical bases development process.

VI. Concluding Remarks

The NRC has initiated a program to revisit the technical bases for the PTS rule and to, potentially, propose a revision to that rule that would significantly reduce the unnecessary level of conservatism in the rule. This effort has been undertaken as a full-participatory activity where active participation by the public and industry in evaluating the changes in the relevant technologies has been sought. Since the completion of the earlier studies, new information has resulted in an improved analytical capability to evaluate PTS events. Results from recent preliminary evaluation are encouraging in trying to substantially reduce the conservatisms in the PTS screening criteria and in developing a comprehensive risk-informed performance based methodology to perform structural integrity analysis of vessel to determine plant risk for potential PTS events, and what potential mitigative actions could be taken to alleviate this risk.

Figure 1: Draft Schedule for 10 CFR 50.61 Pressurized Thermal Shock Screening Criterion Re-evaluation

ID	Task Name	Start	Finish
1	Public Meetings to Identify & Resolve Key Issues	4/1/99	12/31/99
2	Re-assess and Update PTS Risk Acceptance Criterion	4/1/99	6/30/00
3	Identify & Perform Key Revisions in PFM Analysis Code	4/1/99	2/29/00
4	PFM Computer Code Testing & Public Workshop	3/1/00	3/31/00
5	Perform Expert Elicitation for Generic Flaw Distributions	8/2/99	4/7/00
6	Develop Generic Flaw Distributions for RPV Materials	4/10/00	4/28/00
7	Prepare Flaw Distribution Report & Public Presentation	5/1/00	6/30/00
8	Perfom PRA Calcs. for Selected 4 Plants	8/2/99	12/29/00
9	Perform Thermal-Hydraulics (TH) Calcs. for the 4 Plants	8/2/99	12/29/00
10	Perform Probablisitic Fracture Mechanics Calcs for 4 Plants	4/3/00	1/31/01
11	Develop Integrated PTS Risk Estimate in Each of 4 Plants	2/1/01	4/13/01
12	Develop PTS Risk Insights for Entire Polpulation of Plants	4/16/01	5/30/01
13	Peer Review of PTS Tech. Bases Revised Results	6/1/01	6/29/01
14	Develop Proposed Changes to RTpts Screening Criterion	6/1/01	9/28/01
15	Presentations on Proposed Changes in Screening Criterion	10/1/01	10/15/01
16	Finalize Tech. Bases to Revise 10 CFR 50.61 PTS Rule	10/16/01	12/27/01
17	ACRS Presentations on PTS Re-evalution Project	7/14/99	12/6/01



Task		Summary	
Progress		Rolled Up Task	
Milestone		Rolled Up Milestone	
Rolled Up Progress			

Date: December 16, 1999

PTSplanA.mpp

REFERENCES:

[1] SECY 82-465, Pressurized Thermal Shock, U.S. NRC, November 23, 1982.

[2] NUREG/CR-4022, "Pressurized Thermal Shock Evaluation of the Calvert Cliffs Unit 1 Nuclear Power Plant," US NRC, 1985

[3] NUREG/CR-4183, "Pressurized Thermal Shock Evaluation of the H.B. Robinson Unit 2 Nuclear Power Plant," U.S. NRC, 1985

[4] NUREG/CR-3770, "Preliminary Development of an Integrated Approach to the Evaluation of Pressurized Thermal Shock as Applied to the Oconee Unit 1 Nuclear Power Plant," U.S. NRC, 1986.

[5] RG 1.154, "Format and Content of Plant Specific Pressurized Thermal Shock Analysis Reports for Pressurized Water Reactor," U.S. NRC, 1987.

[6] SECY 92-283, "Action Plan to Implement Lessons Learned from the Yankee Rowe Rector Vessel Embrittlement Issue," U.S. NRC, August 14, 1992.

[7] Marshall Committee, "An Assessment of the Integrity of PWR Pressure Vessels," Second Report, Chairman: D. W. Marshall, Published by: U.K. Atomic Energy Authority, March 1982.

[8] Oak Ridge National Laboratory, Technical Letter Report ORNL/NRC/LTR-85/32, "Comparison of Plant-Specific Analyses of Pressurized Thermal Shock (Based on Oconee, Calvert Cliffs, and H. B. Robinson)," D. L. Selby, October 15, 1985.

[9] NUREG/CR-5135, "The Thermal Hydraulics of SBLOCAs Relative to Pressurized Thermal Shock," Theofanous, T.G., LaChance, J.L., Williams, K.A., October 1988

[10] Gary E. Wilson and Brent E. Boyack, "The Role of the PIRT Process in Experiment, Code Development and Code Application Associated with Reactor Safety," Nuclear Engineering and Design, Vol. 186, p. 23-37, 1998.

[11] NUREG/CR-6471, Vol. 1, "Characterization of Flaws in U.S. Reactor Pressure Vessels," U.S. NRC, 1998.

[12] RG 1.99, Rev. 2, "Radiation Embrittlement of Reactor Vessel Material," U.S. NRC, 1988.

[13] NUREG/CR-6551, "Improved Embrittlement Correlation for Reactor Pressure Vessel Steels," U.S. NRC, 1998.

[14] ASME, Proceedings of Pressure Vessel & Piping Conference, "Revisiting the Integrated Pressurized Thermal Shock Studies of an Aging Pressurized Water Reactor," Terry Dickson, Shah Malik, John Bryson and Fredric Simonen, August 2-6, 1999.

[15] RG 1.174, "An Approach for Using PRA in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," U.S. NRC, July 1998.

Developing a Generic Flaw Distribution for Reactor Pressure Vessels

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ABSTRACT

The U.S. Nuclear Regulatory (NRC) is re-evaluating the guidance and criteria in the code of federal regulations (CFR) as it relates to reactor vessel integrity, specifically pressurized thermal shock (PTS). Recent ultrasonic examination (UT) of considerable vessel material at Pacific Northwest National Laboratory (PNNL) and industry experiences with Yankee Rowe have provided the NRC with a better understanding of PTS issues. The reevaluation of PTS will consider a risk-informed approach to the PTS rule and also provide important benefits for licensees considering license renewal.

Pressurized thermal shock transients can lead to reactor vessel failure. These transients have occurred at operating reactors but to date they have not resulted in vessel failure. To properly determine the potential or probability for vessel failure from a PTS event an accurate estimate of fabrication flaws is necessary. The characteristics of the fabrication flaw are inputs to fracture mechanics structural calculations which will determine the probability of vessel failure during a PTS event. Also the results will indicate the sizes and locations of flaws that are most likely to cause failures. This information is also an integral input to the overall pressure vessel safety program.

In order to obtain an accurate estimate of fabrication flaws to address PTS events for all classes of reactors a generic flaw distribution must be developed. An expert elicitation process will be used in conjunction with empirical data from PNNL RPV studies and modeling (RR-PRODIGAL Code) in developing generic flaw distributions.

This paper will demonstrate the important relationship between reactor vessel integrity and flaw distributions in reactor pressure vessel material; discuss the PNNL work to date on developing flaw density and distributions for domestic RPVs; and describe the expert elicitation process that is being planned to verify that a generalized flaw distribution can be properly developed and then to assist in developing a generalized flaw distribution.

BACKGROUND

Reactor Vessel Integrity and Flaw Distributions

The U.S. Nuclear Regulatory (NRC) is re-evaluating the guidance and criteria in the code of federal regulations (CFR) as it relates to reactor vessel integrity, specifically pressurized thermal shock (PTS) which challenges the integrity of a reactor vessel wall. Recent ultrasonic examination (UT) of considerable vessel material at Pacific Northwest National Laboratory (PNNL) and industry experiences with Yankee Rowe have provided the NRC with a better understanding of PTS issues. The reevaluation of PTS will consider a risk-informed approach to the PTS rule and also provide important benefits for licensees considering license renewal.

Pressurized thermal shock (PTS) transients can lead to reactor pressure vessel (RPV) failure. Fortunately, to date these transients have not resulted in reactor pressure vessel failure. To determine the potential or probability for RPV failure from a PTS event an accurate estimate of fabrication flaws is necessary. The estimate of the number, locations and sizes of fabrication flaws in reactor pressure vessel (RPV) welds are important inputs to probabilistic fracture mechanics computer codes such as FAVOR (Dickson 1994) and VISA-II (Simonen et al. 1986) for predicting the failure probabilities of reactor pressure vessels. They are also inputs that are believed to have the greatest levels of uncertainty, which is discussed below in the next section. To reduce this level of uncertainty, the NRC is supporting research to establish a better basis for estimating the distributions of flaws in RPVs.

Fracture Mechanics Calculations

As previously stated, probabilistic fracture mechanics computer codes require accurate estimates of the flaw rates to determine the likelihood of vessel rupture during a PTS event. Fracture mechanics calculations during the 1980s at Oak Ridge National Laboratory (ORNL) estimated PTS related vessel failure probabilities (Selby et al. 1985). These calculations (in support of NRC Regulatory Guide 1.154) concluded that the inputs for flaw densities and size distributions were the largest source of uncertainty in failure probability calculations. The ORNL inputs were based on results of the Marshall Committee Study (1982), and involved a number of approximations and conservative assumptions such as arbitrarily placing all flaws at the inner surface of the vessel.

Given the difficulty of improving on the well-known and extensively used Marshall distribution, little research progress was made until the early 1990s. The literature shows the development of two complimentary approaches. One approach involves the statistical analysis of data from nondestructive inservice inspections of welds. Lance et al. (1992) and Rosinski et al. (1997) described the use of data from inservice inspections (ISI) along with statistically based software (the SAVER code) to develop flaw size and density distributions. Another approach developed by Rolls Royce and Associates in the United Kingdom simulates the population of flaws in multi-pass welds by application of an expert system model based on input from experts in the areas of welding and vessel fabrication (Chapman 1993; Chapman, Khaleel and Simonen 1996). Both approaches have the objective of using the best available data and knowledge to estimate fabrication flaw occurrence rates. Other objectives have been to develop a basis for extrapolating flaw occurrence rates to other reactor pressure vessels which have not been subject to detailed examinations, and to estimate the occurrence rates for flaws that are much larger than the flaws which are observed within the limited volumes of examined vessel material.

Reactor Pressure Vessel Fabrication

The fabrication process of the reactor vessels presents a number of variables that must be considered and some may have significant bearing on the flaw distribution. In the U. S. the RPVs were fabricated primarily by three companies, Combustion Engineering (CE) producing 45%, Babcock and Wilcox (B&W) producing 30%, and Chicago Bridge and Iron (CB&I) producing the remaining 25%. Simplistically, the cylindrical portion of most RPVs in the U. S. was constructed by welding together four shell courses (circumferential weldments). Each shell course was a forged ring or the shell courses were constructed by welding three sections of formed plate resulting in axial weldments. The forged rings assembled as the shell contained no axial welds. The as rolled and tempered plate material was procured and, as part of the acceptance process, the vessel manufacturer would specify a volumetric inspection and inspection extent (such as, 100%) and the acceptable/unacceptable base metal flaw threshold.

The welding process used in the fabrication of the reactor vessels varied with each manufacturer. Three different weld processes were used in assembling the reactor vessels: shielded metal arc welding (SMAW), submerged arc welding (SAW) and electroslag welding (ESW). The manual process of SMAW allows for a greater probability for defects. During the SMAW process, welding is interrupted frequently to change electrodes. The automatic process of SAW is less likely to produce discontinuities as compared to SMAW. Both SMAW and SAW can be used for axial and circumferential welds. Electroslag is an automatic process that can only be used to fabricate axial welds. This process produces less defects but any defects found would be larger than the other two processes.

The shell course plates were formed into arcs and the axial weld preparations were machined. These weld surfaces received a surface inspection prior to welding and then the welds were made. During and following the welding, both surface and volumetric inspections were performed. A cladding such as stainless steel was applied to the inside of the shell course. The formed rings were then stacked and welded to form the cylinder. These circumferential weld preparation surfaces were inspected prior to welding and the welds were subjected to inspections during and following welding. Finally, cladding was applied to the inside of the vessel to cover the newly formed circumferential weld and the clad surface was then inspected.

Shop travelers were created during the fabrication process. The travellers generally documented unacceptable flaws that were detected and documented the details of the repairs. The surface examination techniques most commonly used were either magnetic particle testing (MT) or penetrant testing (PT). The volumetric examinations most commonly used were radiographic testing (RT) employing accelerators to obtain the high energies needed to penetrate the thick section RPV steels and ultrasonic testing (UT). The UT inspections may be conducted from the RPV inside surface, from the outside surface or sometimes inspections were conducted from both surfaces. In many cases all of these techniques were used to assure the quality of the base metal, the weldment and the cladding. One of the goals for the Non Destructive Examination (NDE) employed during fabrication was to insure that no flaws would be detected during preservice inspections that would be found to be unacceptable and require repair. RPV repair in the field is very expensive.

The preservice inspection (PSI) was the base line that was created for the quality of the welds prior to beginning operation. The PSI differs from the fabrication inspections (based on ASME Boiler and Pressure Vessel Section III Code) because the PSI is performed on the RPV after installation using the procedures and flaw acceptance/rejection criteria slated for inservice inspection (ASME Section XI Code). Since most of the RPVs in the U. S. were constructed in the 1960s and 1970s, these RPVs received preservice inspections that have all basically met the same requirements. The data obtained from PSI and ISI can be used to supplement the data on fabrication flaw density and flaw size.

OBTAINING DATA FOR DEVELOPING A FLAW DENSITY AND DISTRIBUTION

As previously discussed until the early 1990's, little research progress has been made to update the Marshall flaw distribution and the results of the Marshall Committee Study (1982). In order to address the NRC's re-evaluation of the PTS criteria and possibly remove some of the approximations and conservative assumptions from earlier ORNL fracture mechanics calculations, an updated flaw distributions must be developed. Literature is available on the development of flaw distributions using the empirical data such as from PNNL (Schuster, Doctor, and Heasler, 1998 and 1999) and the simulated flaw distribution from RR-PRODIGAL Code (Chapman and Simonen 1998).

PNNL is under contract to the NRC to develop a generic flaw density and size distribution for RPVs and has obtained empirical data from RPV material from the three major domestic RPV manufacturers representing four different vessels. As previously stated Combustion Engineering, Babcock and Wilcox, and Chicago Bridge & Iron were fabricators of the majority of U.S vessels.

NDE inspections have been performed on the weldment and some base metal of RPV material from Pressure Vessel Research Users Facility (PVRUF) and Shoreham. The very sensitive SAFT-UT(Synthetic Aperture Focusing Technique for Ultrasonic Examination) was used to detect and characterize flaws in PVRUF and Shoreham RPV material (Doctor et al, 1985). NDE work on the remaining RPV material will be performed in FY2000. A comprehensive documentation of the NDE techniques, methods and data from PVRUF and Shoreham vessel material and a description of the numbers and sizes of the flaws that were detected and sized for the various regions of the vessel are documented two NUREG publications (Schuster, Doctor, Heasler, 1998 and 1999). Table 1 is a chart of the RPV material that is being used for this flaw distribution work.

RPV MATERIAL SELECTED FOR GENERIC FLAW DISTRIBUTION					
	MIDLAND	PVRUF	SHOREHAM	RIVER BEND UNIT 2	HOPE CREEK UNIT 2
MANUFACTURER	B&W	CE	CE	CB&I	CB&I
TYPE	PWR	PWR	BWR	BWR	BWR
METERS OF WELD	4	20	24	15	4
YEARS OF CONSTRUCTION	1968-1974	1976-1981	1968-1974	1974-?	1972-?

RPV MATERIAL SELECTED FOR GENERIC FLAW DISTRIBUTION					
	MIDLAND	PVRUF	SHOREHAM	RIVER BEND UNIT 2	HOPE CREEK UNIT 2
FABRICATION PROCESS	Circ Welds	Circ and Axial Welds	Circ and Axial Welds	Circ and Axial Welds	Circ and Axial Welds
Welds Inspected	Circ Welds	Circ Welds	Circ and Axial Welds	Circ and Axial Welds	Circ and Axial Welds
NDE INSPECTION COMPLETED	Y	Y	Y	N	N

Table 1 RPV Material for Development of Flaw Distribution Work.

A summary of the significant observations from the NDE data of PVRUF and Shoreham that are of significance to fracture mechanics work are as follows:

1. The density of flaws in the PVRUF and Shoreham vessels is significantly greater that predicted by a Marshall Distribution.
2. The cumulative flaw rate of the Shoreham vessel material is approximately three times greater than the PVRUF vessel material.
3. Numerous small flaw indications were found on the fusion surfaces of the structural weld with the base metal on both PVRUF and Shoreham.
4. The largest flaw indications were complex in shape and related to weld repair areas.

EXTENSION OF THE EMPIRICAL DATA TO THE FLEET OF RPVS AND FUTURE WORK

The empirical studies are currently being performed to obtain inclusive data for the database of fabrication flaws in RPVs. The database can be augmented through several different strategies. More RPV material can be inspected and the database expanded. The current plans call for inspections to be conducted on weldment material from the River Bend and the Hope Creek RPVs. Immediate plans also include a more extensive evaluation of base metal. In analyzing the base material that was inspected around the weldments from PVRUF, the flaw density in the base material appears to be at least an order of magnitude smaller than that of the weldments. However, there is so much base metal in a RPV, it is therefore important to know the real flaw density and size distribution functions for base metal in order to conduct meaningful fracture mechanics structural integrity assessments.

The use of the RR-PRODIGAL code shows much promise as a tool to be used to extend the empirical database to the remainder of the RPV fleet. The agreement of the SAFT-UT data with the predictions by the RR-PRODIGAL expert system model, indicates that the number and sizes of flaws in the PVRUF vessel are consistent with estimates made by welding experts based on their extensive knowledge and experience gained from the manufacture of a large number of welds similar to those in the PVRUF vessel. Further work is planned to use NDE inspection results to validate the use of PRODIGAL for the Shoreham, River Bend and Hope Creek RPVs.

There are several ways that Inservice Inspection data can be used to augment the work that is being conducted. In all cases, the inspection system (procedure, equipment and personnel) needs to be quantified in terms of its performance capability for detection and characterization. All ISI data is useful but, as the uncertainty of the inspection system performance is reduced, the more useful the NDE data is in augmenting the database and for use in the extrapolation of the data to the entire fleet of RPVs. Performance results from participation in round robin studies such as Programme for the Inspection of Steel Components (Nichols and Crutzen 1988) or in statistically designed performance demonstrations, help to provide the data to better quantify performance. System performance and the PRODIGAL code and/or the empirical database can be used to estimate what a given inspection should find during an ISI. The significance of the ISI data is that it is the only really valid data on the vessels that are inservice. The comparison between the actual ISI data and estimates of expected flaws, if in agreement, can confirm the credibility of a given inspection, provide confidence that an effective inspection was performed and provide some assurance that a given RPV fabrication was consistent with other vessels.

With the latest flaw density data obtained from the NDE of PVRUF and Shoreham RPV material and the development of RR-PRODIGAL, it is feasible that the panel of experts through expert judgement and elicitation will be able to prove that the Marshall flaw distribution can be replaced with an updated flaw distribution.

EXPERT JUDGEMENT PROCESS

The NRC's Office of Research has committed to revising the PTS acceptance criteria and has decided that the expert judgement process will be used in determining the bases for a generic flaw density and distribution.

The formal use of expert judgement (often referred to as expert opinion) has been extensively applied to a number of major studies in the nuclear probabilistic risk assessment area. Scientific inquiry and decision-making have always relied on expert judgement, but the formal use of expert judgement as a well-documented systematic process is also being used. It has been necessitated by the need to address questions where alternative sources of information are unavailable, less reliable, or too costly. In the case of the development of a generic flaw distribution for domestic RPVs, expert judgement is needed to review, interpret and supplement available information on reactor vessel fabrication processes and reactor vessel flaw distributions. The experts will review work to date by PNNL on empirical data for developing a generic flaw density and distribution for domestic reactor pressure vessels.

The expert judgement process used will have eight steps. These eight steps, which form the basis for the use of expert judgement in the development of the generic flaw distribution, are discussed in the next sections.

Selection of Issues and Experts

The selection of issues and experts is closely related. The initial selection of issues was developed by NRC staff and PNNL staff and used to guide the selection of experts. The experts will review the list of issues and propose additions, deletions, or modifications to the list. There are two ways to organize the experts, either by panels or teams. Regardless of the approach, it is essential that the experts be knowledgeable about the state of the art and be chosen to represent a diversity of backgrounds, with a wide variety of viewpoints (e.g. academic, consulting, commercial, national laboratory, etc.).

For the development of the panel for the generic flaw distribution, the panel was obtained from persons who are experts in one or more of the following areas: ASME Code for Construction, and Inservice Inspection; failure analysis; forgings; metallurgy; non destructive examination (NDE); reactor vessel fabrication; statistics; and welding. The experts were selected on the basis of their recognized expertise in the issue areas, such as demonstrated by their publications, etc.

The preliminary set of issues developed by NRC and PNNL staff are listed below:

1. Clarification of objective, the development of a generic flaw distribution for domestic reactor pressure vessels
2. With three fabricators and different fabrication processes, how did you determine that one distribution is sufficient?
3. What is the significance of the differences between the flaw distribution for each vessel fabricator?
4. What fabrication variables were considered?
5. Did you rank variables by significance?
6. What was your basis for eliminating certain variables?
7. Are there differences between the plate distribution and the weldment distribution?
8. What distribution was determined to be more significant and why?
9. What type/size flaws are of importance and why?
10. Were certain fabrication processes more susceptible to flaws?
11. Were certain welding process more susceptible to flaws?
12. What must be done to create a surface breaking flaw?
13. Where has industry located surface breaking flaws (nuclear and non-nuclear)?
14. Regarding the cladding process, how much is the base metal affected during the cladding process, in terms of under clad cracking and is under clad cracking more prevalent in French vessels?
15. What are the factors, variables or determinants that will have an influence on the distribution of fabrication flaws? Are there more than the list below?

    Base Metal (Plates, forged rings)

        What NDE procedure was used (sensitivity, accept/reject criteria)?

        What are the flaw specifics (type, location, size)

Were flaws surface or embedded?

How many flaws per plate were detected?

Was there a difference for plates in the beltline vs. nozzle shell, etc?

What was the largest flaw detected & repaired?

Was NDE performed on all surfaces of the plates?

Did one surface contain more flaws than another surface?

Welding procedure

Welding materials

Weld design

Repairs (base metal, cladding, weldment)

Cladding

What NDE procedure was used?

What are the flaw characteristics that required repair

What was the location of most flaws?

Describe the repair process

Pre- and Post Hatch

16. Is more data available for naval vessels than for commercial nuclear power plant (NPP) vessels?
17. What was the difference in steel used in NPP vessels and naval vessels?
18. What caused the cracking the head of the Quad Cities vessel in the 1990s?
19. Location of vessel repairs, is there a pattern as to where the repairs are located?
20. Are NDE results of pre-Hatch vessels less reliable and are the vessels more susceptible to flaws than post-Hatch vessels? What effect did the change of NDE of vessel fabrication processes have as there were definitely more repairs? (Hatch History is discussed below)

Prior to 1971/1972 a major discontinuity was discovered in a Hatch vessel nozzle after delivery.

Pre-Hatch era was prior to 1971

Reaction to Hatch era 1971-1975

Stabilizing era after 1975

Prior to the Hatch incident, no UT beyond the basic ASME Sec III was performed. During the reaction era numerous repairs were made because of the dramatic increase in UT requirements so vessels delivered between 1974 and 1977 had an increase repair rate. For vessels delivered after 1977 the repair rate was lower due to improvements n the welding and cladding processes

Prior to the presentation of the issues to the experts and the elicitation training, the issues listed above will be grouped into no more than three categories to ensure a clear understanding by each expert and the elicitation team. It is important to note that responses are needed for each issue whether it is formed as a question or a statement.

Presentation of Issues to the Experts

In addition to providing the experts with a clear statement and a good understanding of the issues, this step provides a mechanism to discuss the state-of the art for each issue. An essential aspect of issue presentation is issue decomposition, which allows the experts to make a series of simpler assessments rather than one overall assessment of a complex issue. This step should be carried out with great care, as the decomposition of an issue can vary by expert and thereby significantly affect its assessment. Care should also be taken to present the issues so as to minimize biases in the expert's assessment.

Elicitation Training

The purpose of the elicitation training is to help the experts learn how to encode their knowledge and beliefs into a quantitative form. Elicitation training can significantly improve the quality of the experts' assessments by avoiding psychological pitfalls that can lead to biased and/or other overconfident assessments. Whenever the training session takes place, it is important that it not be abbreviated due to time pressure. The training should be carried out by a substantive expert who is knowledgeable about the issues to be assessed and an normative expert who is knowledgeable about decision theory and the practice of probability elicitation.

Preparation of Issue Analyses by the Experts

In order to perform a comprehensive issue analyses, the experts should be given sufficient time and resources to analyze all of the issues before the elicitation session. The preparation of issue analyses by the experts may entail support by NRC/PNNL staff e.g performing computer calculations or other requested analyses.

Discussion of Issue Analyses

Before the elicitation session, the experts will be allowed to present the results of their analyses and research. The ensuing discussion can serve to ensure a common understanding of the issues and the data. The final part of this step is to reach agreement on the exact elicitation topics/variables.

Elicitation of the Expert

The elicitation session will be conducted by an elicitation team and should be held immediately following the discussion of the issue analyses. An elicitation team should meet separately with each expert, to avoid pressure to conform and other group dynamics interaction which might occur if the expert judgements were elicited in a group setting. The elicitation team will consist of a substantive expert, a normative expert, and a recorder. The elicitation team will present the issues/ questions to the experts and prepare the results for the substantive and normative experts in this case referred to as a Technical Facilitator Integrator (TFI). The TFI is responsible for aggregating the judgements and community results of the panel of experts. It is also useful to add as a fourth member the person who will prepare the final documentation.

Recomposition and Aggregation of Results

Each expert's elicitations will be recomposed by the normative and substantive experts (TFI) to put them in a form suitable for further analysis.

After recomposition of each expert's elicitation, the results should be aggregated to yield a final assessment for each issue. There are two general classes of aggregation methods, methods that tend to consensus and methods that tend to preserve the variability between the experts. Prior to documentation the expert panel will meet review the results.

Documentation

The final step in the expert judgement process is to document the entire process. Documentation has several purposes. First, it can be used by the experts involved to assure them that their judgements were correctly reflected. Second, it can be used by potential users of the results of the process to enhance their understanding. Third, it can be used by peer reviewers of the process to provide an informed basis for their review. And finally, documentation can be extremely useful to update the analyses, when future research on other vessel material becomes available.

The expert judgement process will be completed by May 2000. Documentation of the process will be published in a NUREG document and a public workshop to present the results of the expert judgement process will be held in late June of 2000..

<u>CONCLUSIONS</u>

The commitment by the NRC's Office of Research to develop a generic flaw distribution for reactor pressure vessels has been positively received by industry and the NRC's Advisory Committee on Reactor Safeguards (ACRS). The development of a generic flaw distribution will be a step forward in possibly removing a source of uncertainty in the fracture mechanics calculations for reactor vessel failure and may result in justification for removal of some of the conservative assumptions regarding reactor pressure vessel integrity guidance in the CFR. The expert judgement process is complex. The recent NDE data from Shoreham and PVRUF material proves that NDE has played and continues to play a very important role in the assessment of the integrity of RPVs and will be a significant topic during the development of the generic flaw distribution.

REFERENCES

Abramson, L.R., et al, 1999. *Uncertainty Evaluation Methods for Waste Package Performance Assessment, NUREG/CR-5639*, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555

ASME Boiler and Pressure Vessel Code. 1998. Published by the American Society of Mechanical Engineers. New York, NY.

Chapman, O.J.V. 1993. "Simulation of Defects in Weld Construction," *Reliability and Risk in Pressure Vessels and Piping*, PVP-Vol. 251, pp 81-89, American Society of Mechanical Engineers, New York.

Chapman, O.J.V., Khaleel, M. A. and F. A. Simonen. 1996. "A Simulation Model for Estimating Probabilities of Defects in Weld", ASME PVP Vol. 323, pp. 375-391, *Fatigue and Fracture - 1996 - Volume 1.*

Chapman, O.J.V. and F.A. Simonen. 1998. *RR-PRODIGAL - Model for Estimating the Probabilities of Defects in Reactor Pressure Vessel Welds*, NUREG/CR-5505, U.S. Nuclear Regulatory Commission, Washington, D.C.

Dickson, T.L. 1994. *FAVOR - Fracture Analysis Code for Nuclear Reactor Pressure Vessels, Preliminary Users Guide*, February 1994 and FAVOR Release 9501, September 1995, Oak Ridge National Laboratory, U.S. Nuclear Regulatory Commission, Public Document Room.

Doctor, S.R., G.J. Schuster, L.D. Reid and T.E. Hall. 1995. Development and Validation of a Real-Time SAFT-UT System of the Inspection of Light Water Reactor Components, NUREG/CR-4583, U.S. Nuclear Regulatory Commission, Washington, D.C.

Lance, J., et al. 1992. "Flaw Distributions and Use of ISI Data in RPV Integrity Evaluations," Edited by M.J. Wittle, K. Iida, F. Ammirato and G.M. Light, ASM 11th International Conference on NDE in the Nuclear and Pressure Vessel Industry, Albuquerque, USA, April 30-May 2, 1992.

Marshall Committee. 1982. *An Assessment of the Integrity of PWR Pressure Vessels*, Second Report by a Study Group under the Chairmanship of D.W. Marshall, published by U.K. Atomic Energy Authority, March, 1982.

Nichols, R.W., and S. Crutzen. 1988. *Ultrasonic Inspection of Heavy Section Steel components, The PISC-II Final Report.* Elsevier Applied Science, New York, NY

Rosinski, S.T., R.P Menish, J.R. Foulds, S. Sanjeev, E.L. Kennedy and D.D. Dedhia. 1997. *SAVER (Statistical Analysis of Vessel Examination Results) A Computer Code for Developing a Flaw Distribution from Vessel Inspection*, SAND94-2454, prepared by Sandia National Laboratory for U.S. Department of Energy.

Schuster, G.J., S.R. Doctor, and P.G. Heasler. 1998. *Characterization of Flaws in U.S. Reactor Pressure Vessels, Density and Distribution of Flaw Indications in PVRUF*, NUREG/CR-6471, Vol. 1, PNNL-11143, U.S. Nuclear Regulatory Commission, Washington, D.C.

Schuster, G.J., S.R. Doctor, and P.G. Heasler. 1999. *Characterization of Flaws in U.S. Reactor Pressure Vessels, Density and Distribution of Flaw Indications in the Shoreham*

Vessel, NUREG/CR-6471, Vol. 3, PNNL-11143, U.S. Nuclear Regulatory Commission, Washington, D.C.

Selby, D.L. et al. 1985. *Pressurized Thermal Shock Evaluation of the H.B. Robinson Nuclear Power Plant*, NUREG/CR-4022, prepared by Oak Ridge National Laboratory for the U.S. Nuclear Regulatory Commission.

Simonen, F. A., K. I. Johnson, A. M. Liebetrau, D. W. Engel, E. P. Simonen. 1986. *VISA-II-A Computer Code for Predicting the Probability of Reactor Pressure Vessel Failure*, NUREG/CR-4486, PNL-5775.

U.S. Nuclear Regulatory Commission. 1987. *Regulatory Guide 1.154, Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors.*

Reactor Vessel Lower Head Failure Experiments and Analyses*

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ABSTRACT

The lower head of the reactor pressure vessel can be subjected to significant thermal and pressure loads in the event of a core meltdown accident; consequently, the possibility exists that the lower head will fail releasing large quantities of core material to the containment. The objectives of this NRC-sponsored program are to characterize the mode, timing, and size of lower head failure (LHF). The program consisted of coordinated experimental and analytical efforts. The present paper provides a concise summary of the experimental and the modeling results of the LHF program.

A Phenomena Identification and Ranking Table (PIRT) was prepared, and a scaling analysis was performed at the start of the program to guide the design and conduct of the experiment program. The experiment program consisted of eight experiments. Each test vessel was a 1/5-scale model of a typical PWR lower head made from prototypic materials (SA533B1 steel). By using geometrical scaling and prototypical material, membrane stress and material behavior were preserved in the experiments. Each experiment was pressurized and heated from the inside until failure occurred.

The test matrix addressed issues of heating patterns, lower head penetrations, lower head weldments, and reactor coolant system (RCS) pressure. Of the eight experiments, seven were performed at a RCS pressure of 10 MPa (nominal membrane stress of 75 MPa) and one experiment was performed at 5 MPa (nominal membrane stress of 37.5 MPa). One replicate test was performed. The key experimental observations are:

- Failure size was typically smaller than the heated region and localized heating led to localized failure;
- Failure typically initiated near locations of high membrane stress (due to variations in wall thickness or temperature or both) and the membrane stress at failure initiation was typically 40% to 60% of yield. Therefore, failure was due to creep;
- Penetrations can lead to premature local weld failure;
- Change of membrane stress (pressure) can result in significant changes in the characteristics and size of local failure; and
- The experiments were highly repeatable and, therefore, amenable to modeling.

The results of the experiments were analyzed using both FEM numerical simulation and Larson-Miller correlation-based system code models. An exhaustive review and analysis of existing

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material database were made, and selected property measurements were made to anchor the diverse material database. A power-law secondary creep constitutive model was implemented in the ABAQUS code. The FEM analyses show good agreement with experimental observations including time to failure and failure location. Vessel failure time calculated from the system code was shown to be in good agreement with experiments. One important lesson learned in the analysis program was the need to develop an accurate constitutive model for the specific material used in the experiment.

Introduction

The present paper provides a concise summary of the experimental and the modeling results of the LHF program at the Sandia National Laboratories (SNL) under the USNRC-sponsored Lower Head Failure (LHF) Program. Detail results can be found in NUREG/CR-5582 (Chu and Pilch et al. 1999). While there were a number of reactor safety related creep rupture investigations (Autrusson and Combescure 1998; Chambers 1989; Devos et al. 1996; Harada, et al. 1997; Maile et al. 1990; Meda et al. 1997; Sievers and Liu 1998; and Strub, C., and J. Devos 1996), the present study was the only one where experiments were performed with prototypical reactor pressure vessel geometry. Recently, Seghal et al. (1998) initiated the FOREVER experiments on thermal and mechanical behavior of reactor pressure vessels under severe accident conditions.

The lower head of a reactor pressure vessel (RPV) can be subjected to significant thermal and pressure loads in the event of a core meltdown accident. Consequently, the possibility exists that creep deformation of the lower head will eventually lead to the failure of the reactor pressure vessel, releasing large quantities of core material to the containment.

The initial motivation for the LHF Program was the inability of the state-of-the-art analyses in the OECD/NEA TMI-II Vessel Integrity Program (VIP) to simulate the known outcome of the TMI accident: the TMI vessel did not fail (OECD 1994 and Wolf et al. 1994). However, of more current interest is the increased need to analyze existing and next generation reactors from the perspective of accident mitigation and accident management. The characterization of vessel creep deformation, as a precursor to water ingression leading to in-vessel cooling, will assume increased importance (Hodges et al. 1991 and Theofanous et al. 1996). Vessel failure is of importance in accident assessment because the lower head failure defines the initial conditions of all ex-vessel events. Understanding vessel deformation and failure is also important in the assessment of accident management and accident mitigation strategies. The goals of the LHF program were to experimentally characterize the deformation and failure of the reactor vessel lower head, and to develop the analytical capability to predict the deformation and failure of the lower head based on the experimental observations.

Experiment Design

A Phenomena Identification and Ranking Table (PIRT) assessment and scaling analysis were performed to guide the design of the experimental program. Scaling considerations require: (1)

the test vessel to be geometrically scaled to a typical PWR reactor to preserve the membrane stress, (2) SA533B1, the prototypical material for US PWRs, to be used to preserve the material behavior, and (3) the applied heat flux to the lower head to be scaled by the geometrical scale to preserve the creep and failure time.

The experiments were performed using 1-to-4.85 linear scale models of a typical PWR lower head. To be useful for model validation, the experiments were designed with well-characterized initial and boundary conditions and with sufficiently detailed measurements of temperature, pressure, and displacement histories. Maps of the vessel shape (including wall thickness) are obtained pre- and posttest for comparison with model calculations. Since the heat flux was decreased by the scale factor to preserve heatup timing, the throughwall temperature differential is decreased by the square of the scaling factor. As a result, the throughwall temperature differences in the present LHF experiments are typically about 10 K.

The experimental program consisted of a series of eight experiments. The test matrix is given in Table 1 to facilitate the discussion of the experimental results. The test matrix was designed to examine the effects of spatial temperature/heat flux distribution, pressure, and reactor vessel structure elements and construction features on RPV deformation and failure. Three temperature distributions were used: uniform, center-peaked, and side-peaked. A uniform temperature distribution can be considered as a reference case, which might occur in those scenarios where core melt gradually relocates to the lower head. Center-peaked distributions are reminiscent of the hot spot in TMI. Side-peaked distributions correspond to the side-peaked heat flux distribution of a convecting molten pool.

Table 1. Test Matrix of LHF Experiments

Tests	Heat Flux Distribution	Structure Elements	Pressure
LHF-1	Uniform	-----	10 MPa
LHF-2	Center-peaked	-----	10 MPa
LHF-3	Side-peaked	-----	10 MPa
LHF-4	Uniform	Penetrations	10 MPa
LHF-5	Side-peaked	Penetrations	10 MPa with Transient
LHF-6	Uniform	Weldment	10 MPa
LHF-7	Uniform	-----	5 MPa
LHF-8	Side-peaked	-----	10 MPa

Experiment Results

All of the experiments were conducted at a testing pressure of 10 MPa except LHF-7, which had a testing pressure of 5 MPa. A pictorial summary of all eight LHF experiments is presented in Figure1. Table 2 summarizes the test conditions and key results of all the experiments.

The first three experiments (LHF-1-uniform, LHF-2-center-peaked, and LHF3-side-peaked) examined the effect of heat/flux temperature distribution. The results of the first three experiments show fairly consistent temperatures for the onset of creep (935 K to 980 K) and vessel failure (1006 K to 1038 K). The time to failure was typically tens of minutes. The vessel deformed with an ever-increasing rate toward failure. Severe necking was observed at all failure sites with local strain at the failure as large as 200%. The overall vessel deformation was in the range of 10% to 30%. The morphology of the failure surface suggested that the initial failure was tensile in nature and the failure propagated by shear. These general features were found to be common to all the experiments performed at 10 MPa. With local heating, the location of significant vessel deformation and eventual failure coincided with the location of peak temperature. The effect was quite dramatic in LHF-3. With side-peaked heating characteristic of molten pool convection, the vessel failed with a limited, straight, and distinct latitudinal (circumferential) rip following the locus of the peak temperature. The rip spanned only 75° of the full 360° circumference. The distinct latitudinal rip suggested the geometrical potential for circumferential tearing of the lower head, but the failure propagation was limited.

LHF-4 was a replicate of LHF-1 except for thirty scaled penetrations installed in the vessel bottom. The vessel failed prematurely as a result of penetration weld failure. The diameter of the penetration through-holes increased by as much as a factor of two. Consequently, the weld (and not differential thermal expansion as postulated by Rempe et al. 1993 and Rempe et al. 1994) is the only mechanism holding the penetration in place. Analyses of penetration failures should focus on the stress state in the weld, as induced by global deformations.

LHF-5 was intended as a replicate for LHF-3 except for the nine penetrations installed in the vessel bottom. An unplanned re-pressurization transient from 7.7 MPa to 10 MPa occurred while the LHF-5 vessel temperature was just under 1000 K, a temperature above the previously observed temperature for the onset of creep. The vessel responded immediately with increasing rate of deformation. LHF-5 failed catastrophically with the circumferential tearing of the lower head along the locus of peak temperature. A comparison between LHF-3 and LHF-5 suggested that the repressurization transient at elevated vessel temperatures most likely contributed to the circumferential tearing of the vessel.

LHF-6 was designed to investigate the effect of weldment on vessel deformation and failure. The vessel had one latitudinal weld and four longitudinal welds. With uniform heating, the vessel weldment was not challenged significantly. As a result, LHF-6 was essentially an exact replicate of LHF-1. The results of the two experiments were identical not only in terms of general features but also in specific details such as local deformation history and post-test vessel shape. The remarkable consistency between the LHF-1 and LHF-6 results suggest that the RPV creep rupture process is well behaved and therefore amenable to modeling.

LHF-7 was a replicate of LHF-1, except that the testing pressure was reduced from 10 MPa to 5 MPa. The overall deformation of LHF-7 was comparable to that of LHF-1, but the details of vessel failure was rather different. As expected, lowering the driving pressure had the effect of elevating the temperature for the onset of creep and the temperature for vessel failure. The temperatures for the onset of creep and vessel failure for LHF-7 were 992 K and 1200 K, respectively, which are significantly higher than the corresponding temperatures in the 10 MPa experiments. There were significant differences in the failure characteristics. While all LHF experiments at 10 MPa resulted in severe necking with thickness reduction of a factor of ten, the corresponding thickness reduction for LHF-7 was only a factor of two. In this respect, lower pressure (membrane stress) failure might be more amenable to modeling because it does not involve the severe area reduction prior to vessel failure.

LHF-8 was designed as a replicate of LFH-3; however, as performed, LHF-8 differed from LHF-3 in some specific details. The location of the peak temperature differed by 10^{o} latitude, and the peak-to-vessel-bottom temperature differential for LHF-8 was approximately twice that of LHF-3. However, LHF-8 replicated all of the important features of LHF-3. LHF-8 failed with a limited rip along the locus of the peak temperature, and the overall vessel deformation, the thickness of the necking zone, and the temperatures for the onset of creep and vessel failure are all similar to LHF-3. The consistency between LHF-3 and LHF-8 tests results suggests again that key test observations are repeatable when similar initial conditions are applied in the tests.

An assessment of the location of failure and site for failure initiation can be made by considering the entire LHF experimental database. Globally, for uniform temperature failure typically occur in regions of reduced wall thickness; the failure site for LHF-1 is a prime example. For cases where there is localized heating, failure always occurs in regions of maximum temperature; examples are the failure in the hot spot for LHF-2 and the latitudinal rips in LHF-3, LHF-5, and LHF-8. It appears that failure is attracted to regions of weakness (reduced thickness or elevated temperature). The contrast in load-carrying capability need not be very large. The failure region in LHF-1 was less than 5% thinner than the surrounding region. The tendency for failure to seek out the region of weakness also plays a role in determining the local site of failure initiation. For example, in LHF-6, the vessel was uniformly heated, and failure occurred in a latitude region of reduced wall thickness. Within that latitude range, the largest deformation occurred in the region of the highest temperature.

The results of the eight LHF experiments met nearly all the specific objectives of each experiment. Taken as a whole, the LHF experiments fulfilled the objectives of characterizing the mode, timing, and size of lower head failure. The data developed was instrumental in the assessment of SAC (Severe Accident Code) and FEM (Finite Element Code) models of RPV lower head failure.

Perhaps one issue that was not fully resolved was the effect of weldment. Performing simple tensile and creep experiments with weld samples will go a long way toward resolving this issue.

Assessment of Material Property Database

The experimental results were used in the assessment of SAC and FEM models. The experimental results play two roles: one is to provide the data for model assessment and validation, and the other is to guide the thought process in determining parameters or processes of importance.

The first step in creep modeling was to develop a consistent material database. The existing database for tensile properties (elastic modulus, yield stress, plastic modulus, and true ultimate stress) was critically reviewed and correlated. The tensile property database reasonably spans the temperature range of interest to severe accidents. Simple three-parameter curvefits were developed for each tensile property and for two temperature ranges, $T \leq 950$ K and $T > 950$ K, which is not associated with the austenitic/ferritic phase transition temperature. The transition temperature was dictated by a need to have an accurate curvefit in the temperature range of interest to severe accidents. Uncertainties (expressed as factors multiplied by the nominal value) were quantified for each material property curve fit. For the yield stress, the correlation is good to a factor of 1.19 (and 0.84) for plus and minus one standard deviation, respectively.

The existing creep database was also critically reviewed and correlated. The creep database reasonably spans the range of temperatures and stresses of interest to severe accident analyses. We concluded that the existing database could not support the development of a unified viscoplastic constitutive law. A modified power law function was proposed for the creep constitutive law. A three-parameter creep constitutive law was fit to the relevant database for two temperature ranges ($T \leq 1050$ K and $T > 1050$ K) defined by the austenitic/ferritic phase transition that occurs between 1000 K and 1100 K. Uncertainties were quantified for the minimum creep rate, which is predicted to a factor of 1.68 (and 0.594) for plus and minus one standard deviation, respectively. As implemented in ABAQUS (i.e., interpreting the stress as the code-calculated true stress), the creep constitutive law reproduces the creep database with adequate representation of uncertainties. We conclude that "tertiary" creep is predominately a response of the material to area reduction during straining; consequently, tertiary creep need not be modeled as a fundamental property of the material that requires explicit representation beyond what the code already calculates.

The existing database for time-to-failure (assuming creep dominates) as a function of constant load and constant temperature was critically reviewed and correlated. The Larson-Miller Parameter Correlation (LMP) was fitted to the existing database. Uncertainties in correlating the time-to-failure were unacceptably large when the complete database was included and a single correlation was used for all temperatures. The uncertainties were greatly reduced when the database was restricted ($t_{fail} < 720$ hr) to conditions representative of severe accident analyses. Uncertainties were also reduced significantly if the correlation was developed separately for two temperature ranges ($T \leq 1050$ K and $T > 1050$ K), as dictated by the approximate austenitic/ferritic phase transition in the steel. An optimal correlation was developed by restricting the database and by splitting the correlation at the phase transition temperature. For the optimal correlation, time-to-failure was correlated to a factor of 2.28 (and 0.439) for plus and minus one standard deviation, respectively.

Because of the large scatter between different data sets in the existing database, we made supplementary material property measurements to ascertain where properties for the test vessel material may fall within the spread of the existing material property database. The property tests results show that:

1. The supplemental measurements of the elastic modulus are consistently lower than the recommended correlation (based on the existing database) and well outside the scatter of the existing database.

2. The supplemental measurements of the yield stress and engineering ultimate strength are consistently higher than the recommended correlations (based on the existing database); however, the supplemental measurements are within ~$2\sigma_{err}$ of the existing database.

3. The supplemental measurements of the creep rate are consistently higher than the correlation based on the existing database for the 973 K temperature. The supplemental measurements of creep rates are as much as 4.3 times higher (2.8 standard deviations) than the recommended creep correlation (based on the existing database). Supplemental measurements of the creep rate at 1173 K are more consistent with our recommended correlation and well within the scatter of the existing database.

4. The material used in the LHF tests generally behaves in an isotropic manner.

5. The supplemental measurements of material properties are generally reproducible with a scatter similar to that exhibited by the existing database.

The limited material property testing performed here is not adequate to fully characterize the material used in the LHF tests for the full range of temperatures of interest. We recommend that a more complete characterization of the material used in the completed LHF tests be performed to support any formal benchmarking activities that possibly might be organized around the completed LHF tests. We further recommend that any possible future LHF-like testing be accompanied by a material property testing program to reasonably characterize the actual material used in the program. Replicate tests are an important element of a material testing program, and it may be necessary to pre-qualify the test specimens to understand why replicate creep tests sometimes yield significantly different results.

The supplementary property measurements also provided important physical insights in the interpretation of the (5 MPa) LHF-7 experiment. The failure temperature for LHF-7 was higher than that observed in all the other experiments, and the two sets of failure temperatures straddle the phase transition region. LHF-7 did not exhibit the severe necking observed in the other experiments. Physical examination of material testing samples above and below the phase transition region confirmed the differences in failure characteristics observed in the LHF experiments. The difference in creep (and tensile) failure behavior also supports the use of different creep correlations above and below phase transition. In LHF-7, the direction of failure propagation was longitudinal, whereas failure propagation in the 10 MPa experiments was generally latitudinal. The supplementary measurements showed the LHF steel to behave isotropically; thus non-isotropy appears not to be important in failure propagation.

Assessment of Severe Accident Code (SAC) Methodologies

Simple "engineering" methodologies associated with creep-based models and strength-based models were evaluated. These methodologies are commonly employed in Severe Accident Codes (SACs, e.g., SCDAP/RELAP5, MELCOR, and MAAP4) that model the full sequence of events that occur in a core melt accident. Within the uncertainties in these methodologies, the creep-based methods nominally correlate both the time for onset of creep and failure times observed in the LHF tests. The key dependency on system pressure is captured by the creep-based methodology. Strength-based models consistently overpredict the onset time for large deformations and the failure time. The key dependence on system pressure is not captured by the strength-based models. Consequently, we conclude that creep, not plastic deformation, is the dominant mechanism leading to failure in LHF tests.

We recommend use of the Huddleston effective stress (Huddleston 1985 and Rempe et al. 1993) in conjunction with the creep based methods. Using the simple membrane stress often gives reasonable results with a consistent bias to overpredict failure times. Using the Huddleston effective stress gives better and more consistent results. We acknowledge that further work is needed to add rigor to this recommendation.

Creep based methods, using the vessel diameter in stress evaluations, can significantly underpredict the vessel failure time for localized heating patterns producing a TMI-like hot spot. We tentatively recommend using the hot-spot diameter in these situations, as this brought LHF-2 predictions into better agreement with observations. A summary comparison of the predicted and measured failure times is shown in Figure 2.

Creep-based methods should be based on the local latitudinal temperatures. The failure location can be reasonably predicted when large non-uniformities in latitudinal temperatures are present (i.e., situations with side-peaked heating or hot spots). The failure location defines an upper bound to the possible hole size if complete circumferential tearing is assumed (which was observed in only one LHF test).

The creep-based models employ a lifetime rule in conjunction with a Larson-Miller-Parameter (LMP) time-to-failure correlation. Stochastic uncertainties in the LMP correlation can introduce uncertainties into the evaluation of vessel failure times. These uncertainties should be reflected in any evaluations concerning the relative failure timing of reactor coolant system components.

The semi-empirical lifetime rule (Dosanjh and Pilch 1991) is used to bridge the gap between the LMP database (constant temperature and constant load) and reactor applications where temperature and pressure states are often time varying. The overall methodology using the lifetime rule in conjunction with the LMP correlation gave reasonable results for the temperature transients and limited pressure transients exhibited by the existing database. We cannot judge if the lifetime rule introduces significant additional uncertainty, beyond that associated with the LMP correlation, into the evaluation of vessel failure times. The technical basis for the lifetime rule precludes its direct use (for the purpose of defining nodal failures) in a FEM code that supplies a true stress (rather than an engineering stress) to the LMP correlation.

The existing LHF database is inadequate for a complete assessment of creep-based methodologies. Notably lacking are data with large through-wall temperature gradients resulting in significant redistribution of internal stress from the hot inside surface to the cooler outside surface of the lower head.

Assessment of Finite Element Models (FEM)

Finite Element Model (FEM) methods were assessed against the LHF-1, LHF-3, LHF-5, and LHF-7 tests. Axisymmetric computer models for the LHF tests accurately specified the vessel geometry (including local vessel thickness), the boundary conditions, and the temperature history throughout the vessel. Large vessel deformations and failure were consistently delayed relative to experiment results; however, the failure location was reasonably predicted. Sensitivity studies showed that the creep rate was the dominant factor controlling the onset of large deformations and the deformation history leading up to failure. Additional sensitivity studies showed that the four LHF tests could be reasonably predicted if the creep rate was increased by a factor of ≈4-5. These observations are well illustrated in the FEM simulation of LHF-1, see Figure 3.

A supplemental material property testing program was conducted using material from the LHF tests. Limited creep testing using specimens taken from the LHF tests suggests that the creep rate is ≈2–4 times faster than our creep correlations would predict using the existing database, consistent with the results of the finite element model sensitivity studies.

It seems reasonable, that deviations between predictions and experiment results could be resolved if a more complete material property program were conducted with material specific to the tests. The supplemental material testing that was performed as part of this program is inadequate to fully characterize the materials used in the LHF tests because property tests were only conducted at two selected temperatures, one on either side of the phase transition temperature.

The model recommended here differs from the OECD/TMI/VIP model in four significant ways:

1. Our assessments were performed with the finite element code, ABAQUS (1998), using a two-dimensional axisymmetric code that is commercially available to the international community.

2. Our creep law is based on a more systematic and selective use of the creep database.

3. We recommend against the use of the OECD/TMI/VIP damage law in the context of finite element calculations.

4. We recommend against the use of an independent correlation for tertiary creep.

It is outside the scope of this report to reanalyze TMI-II; however, the impact of our modeling recommendations would be to significantly slow down the evolution of large deformations and delay failure relative to the OECD/TMI-II predictions. It remains to be demonstrated, however, that the models recommended here would actually produce TMI-II predictions that are in agreement with the intact end-state of the TMI-II lower head.

Conclusion and Potential Future Activities

The LHF experiments fulfilled the objectives of characterizing the mode, timing, and size of lower head failure (LHF). The data were used to evaluate the modeling approaches used in the Severe Accident Codes. Specific recommendations for improvements were made. A constitutive law for creep was developed and implemented in the ABAQUS finite element structural mechanics code. The model differs in significant ways from those used in the OECD/TMI/VIP. Creep processes clearly control vessel failure.

The completed LHF program has formed the foundation for understanding deformation and failure of a RPV lower head during a severe accident. The database, however, is still incomplete. Potential follow-on activities should focus on (1) more prototypic temperature drops across the vessel so that stress redistribution effects can be captured, and (2) system pressures (≈2 MPa) and pressure transients that are representative of managed accidents with intentional depressurization. A round-robin activity would be helpful in assessing international modeling approaches on a common basis. A more quantitative assessment of the models requires more complete material property testing program to better characterize the materials used in the tests. It is expected that the completed LHF program, in conjunction with these recommended activities, will bring closure to the issues of lower head structural response in a severe accident. The current OECD Lower Head Failure (OLHF) program contains major elements of the above outline program except the round-robin activity.

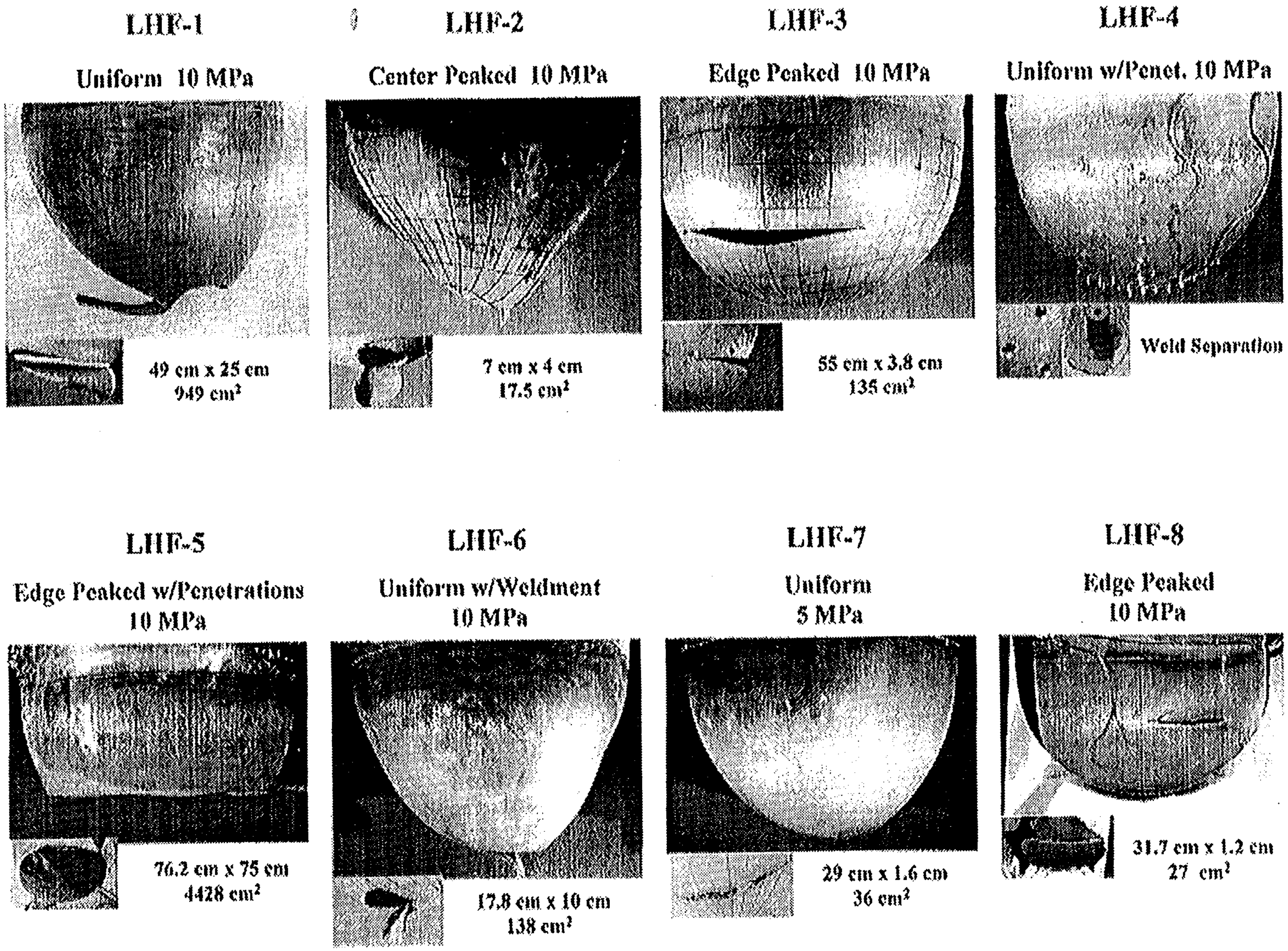

Figure 1. Pictorial summary of the completed LHF tests sponsored by the USNRC

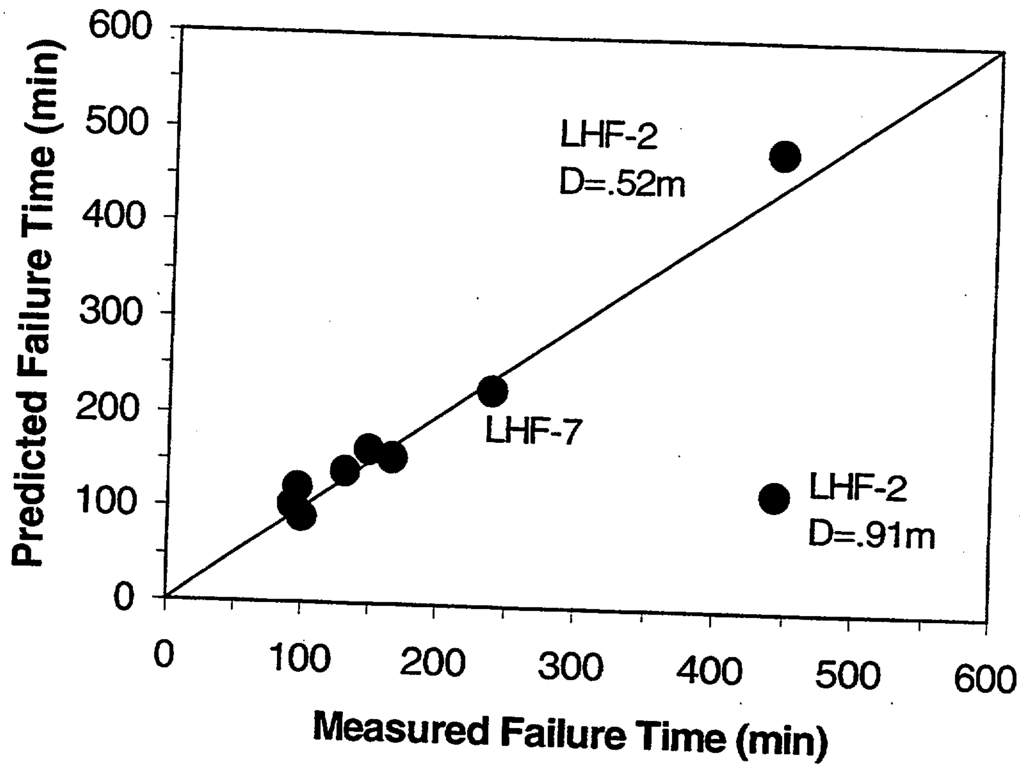

Figure 2. Comparison of predicted and measured failure times using the Huddleston effective stress

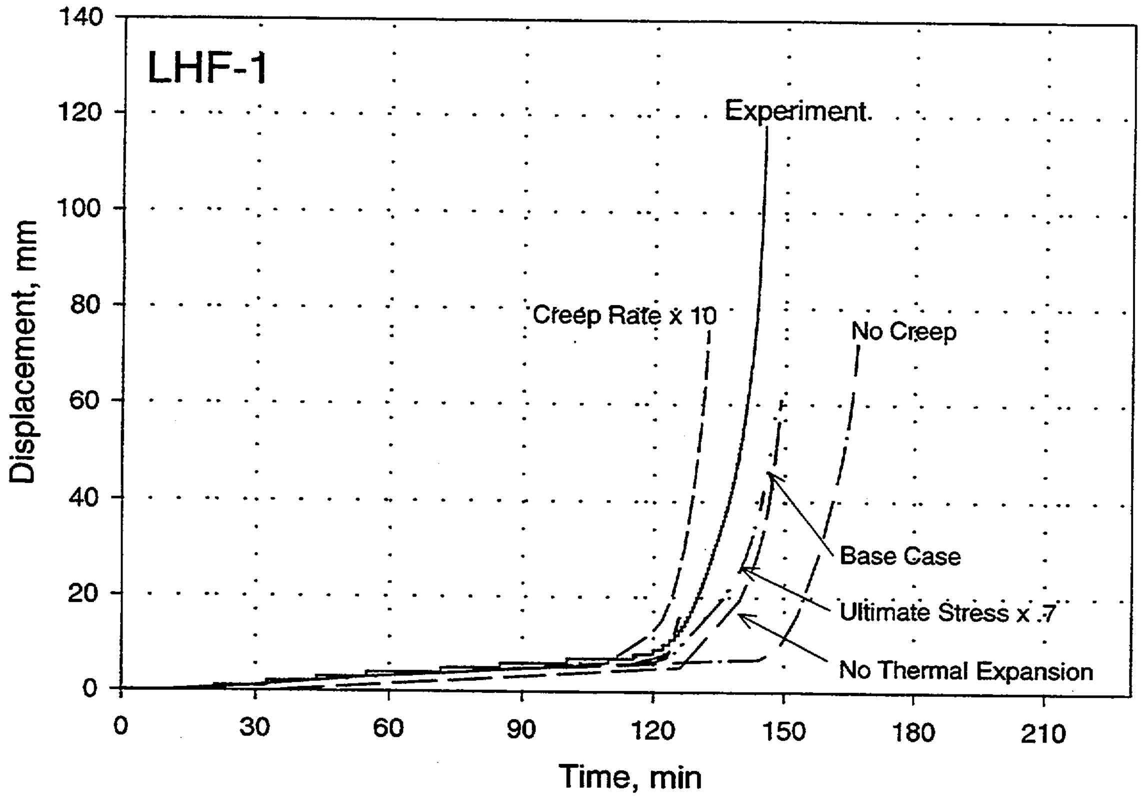

Figure 3. Comparison of 90° Vertical Displacement Predictions and Sensitivities with Experiment Data for LHF-1

Table 2. Summary of LHF experimental results

Test	Heat Flux Distrib.	Test Press. (MPa)	T_{in}(K)	T_f (K)	RPV Features	Failure size/initiation location overall strain/Wall thickness at failure
LHF-1	Uniform	10	935	1038	None	49 cm × 25 cm oval, 1.66m FSE /66° 33%/3 mm
LHF-2	Center Peaked	10	958	1010	None	4 cm ×7 cm oval, 0.23m FSE /77° 34%/3 mm
LHF-3	Side peaked	10	980	1006	None	3.8 cm wide by 55 cm tear, 0.63m FSE/33.5° 11%/3 mm
LHF-4	Uniform	10	949	977	30 scaled penetrations between of 55° and 90° (bottom center) latitude	Penetration weld failure at weld/base-metal interface with less than 1 mm separation. Holes for penetration doubled in size.
LHF-5	Side peaked	7.7-10	997	1114	9 scaled penetrations between 41° and 80° latitude	Complete circumferential tearing, 3.58m FSE
LHF-6	Uniform	10	949	1052	vessel with typical welded construction consisting of a bottom dish and a 4-segment upper torus.	17.8 cm × 10 cm oval, 0.63 m FSE /74° 29%/4 mm
LHF-7	Uniform	5	992	1200	None	Latitudinal rip 1.6 mm wide and 29 cm long, 0.32m FSE /30%/13 mm
LHF-8	Side Peaked	10	967	1041	None	1.2 cm wide by 31.7 cm tear, 0.28m FSE/25° 10%/5mm
T_{in} = Temperature for creep initiation, T_f = Vessel failure temperature FSE=full scale equivalent hole diameter						

References

ABAQUS, "ABAQUS/Standard Users Manual," Hibbit, Karlsson, Sorensen, Inc., 1080 Main St., Pawtucket, RI, 02860-18-17, 1998.

Autrusson, B., and A. Combescure, "Lower Head Thermo-Mechanical Behaviour," *Proceedings of the OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability*, Garching, Germany, March 3-6 1998.

Chambers, R.S., "A Finite Element Analysis of a Reactor Pressure Vessel During a Severe Accident," NUREG/CR-5046, February 1989.

Chu, T.Y., M.M. Pilch, and J.H. Bentz, J.S Ludwigsen, W-Y Lu and L.L. Humphries "Lower Head Failure Experiments and Analyses," NUREG/CR-5582, SAND98-2047, Sandia National Laboratories, Albuquerque, NM, February 1999.

Devos, J.M., C. Sainte Catherine, and C. Strub, "RUPTHER: a Programme Devoted to Modelling the Failure of the RPV Lower Head, *ASME PVP Conference Montreal,* 1996.

Dosanjh, S.S, and M. Pilch, "Lower Head Creep Rupture Sensitivity Studies," *Nuclear Science and Engineering*, Vol. 108:2, pp. 172-183, 1991.

Harada, Y., A. Maeda, Y. Maruyama, H. Shibazaki, A. Hidaka, and J. Sugimoto, "Metallurgical Examination of Piping Failed at High Pressure and High Temperature in WIND Project," *Presented at the SARJ-97*, Yokohama, Japan, October 1997.

Hodges, S.A., J.C. Cleveland, and T.S. Kress, "External Flooding of a BWR Reactor Vessel as a Late Accident Mitigation Strategy," *Letter Report*, ORNL/NRC/LTR-91/9, Oak Ridge National Laboratory, Oak Ridge, TN, August 27, 1991.

Huddleston, R.L., "An Improved Multiaxial Creep-Rupture Strength Criterion," *Journal of Pressure Vessel Technology*, Vol. 107, pp. 421-429, November 1985.

Maile, K., A. Klenk, V. Obst, and D. Sturm, "Load Carrying Behaviour of the Primary System of PWRs for Loads Beyond the Design Limits. Part 2: Creep and Failure Behaviour of a Piping Section Under Internal Pressure and High Temperature," *Nuclear Engineering and Design 119,* pp. 131-137, 1990.

Maeda, A., Y. Maruyama, Y. Harada, H. Shibazaki, N. Nakamura, A. Hidako, and J. Sugimoto, "Studies on Reactor Piping Integrity during Severe Accident in WIND Project," presented at the SARJ-97, Yokohama, Japan, October 1997.

OECD, "Three Mile Island Reactor Pressure Vessel Investigation Project," *Proceedings of an open forum sponsored by the OECD NEA and the USNRC*, 1994.

Rempe, J.L., G.L. Thinnes, and S.A. Chavez, "A Simpler Approach for Predicting LWR Vessel Failure During Severe Accident Conditions," *Nuclear Engineering and Design 148,* pp. 365-383, 1994.

Rempe, et al., "Light Water Reactor Lower Head Failure Analysis," NUREG/CR-5642, EGG-2618, Idaho National Engineering Laboratory, 1993.

Rempe, J.L., and T.J. Walker, "Overview of NRC Sponsored LWR Lower Head Failure Research Program," *Nuclear Engineering and Design 148,* pp. 327-341, 1994.

Sehgal, B.R., R.R. Nourgaliev, T.N. Dinh, A. Karbojian, J.A. Green, and V. A. Bui, "FOREVER Experiments on Thermal and Mechanical Behavior of a Reactor Pressure Vessel during a Severe Accident," *Proceedings of the OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability,* Garching, Germany, March 3-6 1998.

Sievers, J., and X. Liu, "Parametric Studies on Creep Behavior of a Reactor Pressure Vessel Lower Head," *Proceedings of the OECD/CSNI Workshop on In-Vessel Core Debris Retention and Coolability,* Garching, Germany, March 3-6 1998.

Strub, C., and J. Devos, "Modelling of Creep Rupture Experiments Using a Coupled Damage – Viscoplasticity Model," *ASME PVP Conference*, Montreal, July 1996.

Theofanous, T.G., et al., "In-vessel Coolability and Retention of a Core Melt," DOE/ID-10460, 1996.

Wolf, J.R., et al., TMI-2 Vessel Investigation Project Integration Report, NUREG/CR-6197, TMI Y(93)EG10, EGG-2734, Idaho National Engineering Laboratory, 1994.

Reactor Pressure Vessel Embrittlement: The Road Ahead

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ABSTRACT

Current methods of reactor pressure vessel transition temperature shift predictions, formulated from physically based models of defect evolution and embritttlement micromechanics, are generally accurate. However, they may not fully account for all potentially significant variables, such as product form and attenuation, and phenomena occurring late in service life. The master curve method appears to be an empirically successful approach for measuring transition $K_{Jc}(T)$ data, but the underlying assumptions need to be rigorously tested. Vessel sampling programs offer great promise to resolve many open issues, but they must address the issue of surrogate materials and have a proper framework of supporting information.

INTRODUCTION

The last decade has seen remarkable progress in developing a mechanistic understanding of irradiation embrittlement of reactor pressure vessel (RPV) steels. This understanding has been exploited in formulating robust, physically based and statistically calibrated models of Charpy V-Notch indexed transition temperature shifts. These semi-empirical models account for key embrittlement variables and variable interactions, including the effects of copper (Cu), nickel (Ni), phosphorous (P), fluence (ϕt), flux (ϕ) and irradiation temperature (T_i). Models of evolution of nanoscale precipitates rich in copper, manganese and nickel are quantitatively consistent with experimental observations of the complex interplay between these elements and other embrittlement variables. The models also rationalize other effects, such as those associated with post weld heat treatment and many aspects of the interactive flux-composition-temperature dependence of embrittlement. Models have been extended to treat post irradiation annealing and re-embrittlement based on tracking the fate of key alloy constituents and defects. Finally, revolutionary advances have been made in directly measuring fracture toughness, using a relatively small number of relatively small specimens based on the master curve (MC) concept. In this paper, we very briefly review this progress and highlight some outstanding issues and opportunities. The focus will be on establishing mean and bounding fracture toughness-temperature $K_{Jc}(T)$ curves, not on other important integrity issues like pressurized thermal shock. The viewpoints are ours; they are subjective and are not meant to represent any position of the Nuclear Regulatory Commission.

EMBRITTLEMENT MODELING AND PREDICTIONS

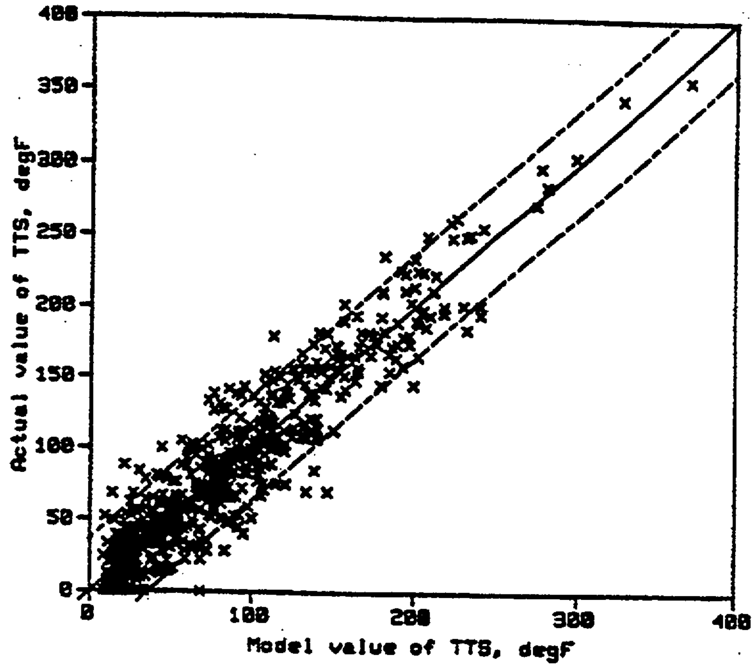
Tremendous progress has been made in developing physically based, statistically calibrated models of the shift in transition temperature ΔT indexed by Charpy impact energies at the 41J level.[1,2] These models account for the contributions of a number of key variables including Cu, Ni and P contents, ϕ, ϕt, and T_i. Similar success has been obtained in modeling recovery of ΔT by post irradiation annealing (PIA)[3,4], again accounting for key metallurgical and irradiation variables as well as annealing times and temperatures. Most recently, some initial success has been achieved in applying a similar approach to modeling re-embrittlement after PIA [4,5]. As shown in Figure 1, the standard errors achieved in the model predictions are generally less than about 13°C, which is within measurement uncertainties.

The physical basis underlying the models is the evolution of several populations of nanoscale features in reactor pressure vessel steels during irradiation. This evolution and the nature of the features are linked to the key embrittlement variables and how they mediate embrittlement through the micromechanics of ΔT. The features can be grouped into three classes. The dominant features in highly embrittled steels are copper-rich precipitates (CRPs) or copper-catalyzed, manganese-nickel rich precipitates (MNPs). In addition, two types of matrix defects evolve: those that are thermally stable (UMDs) and those that are stable (SMFs) at typical reactor pressure vessel operating temperatures. These features evolve primarily as a consequence of radiation enhanced diffusion (RED) and defect clustering, and their evolution can be modeled in terms of these processes. The features lead to increases in the yield stress $\Delta\sigma_y$ by serving as obstacles to dislocation motion, and this hardening mediates the transition temperature shift. This relationship between microstructure, $\Delta\sigma_y$ and ΔT can be modeled by a combination of the micromechanics of dislocation-barrier interaction, computer simulation, and analysis of data trends. The resulting models can be calibrated by data obtained from fundamental systematic studies, generally obtained with the use of research reactors, as well as power reactor surveillance data [2].

In the latter case, the physical models are used to guide the formulation of mathematical constructs to correlate and statistically fit a large data base, such as the Power Reactor Engineering Data Base (PREDB), and to discriminate between statistically equivalent fits.[1,3] The physical models and fundamental data base (now comprising several thousand data points) are quantitatively consistent with the PREDB statistical correlations and provide independent confirmation of the validity and robustness of these correlations.

The success of the resulting models in correlating data bases with a fairly wide range of metallurgical and irradiation variables demonstrates that the separate and combined effects of Cu, Ni, P, Mn, ϕ, ϕt, T_i, post weld heat treatment and annealing times and temperatures are reasonably understood. They have provided early warnings of potential technical surprises – e.g., the contribution of "late blooming" phases such as MNPs in high Ni steels to embrittlement; and they have enabled the assessment of outliers in the data base as well as contradictory observations. However, the models and understanding are not fully quantitative, and do not treat all potentially significant variables and issues. A summary of these issues is discussed below.

a)



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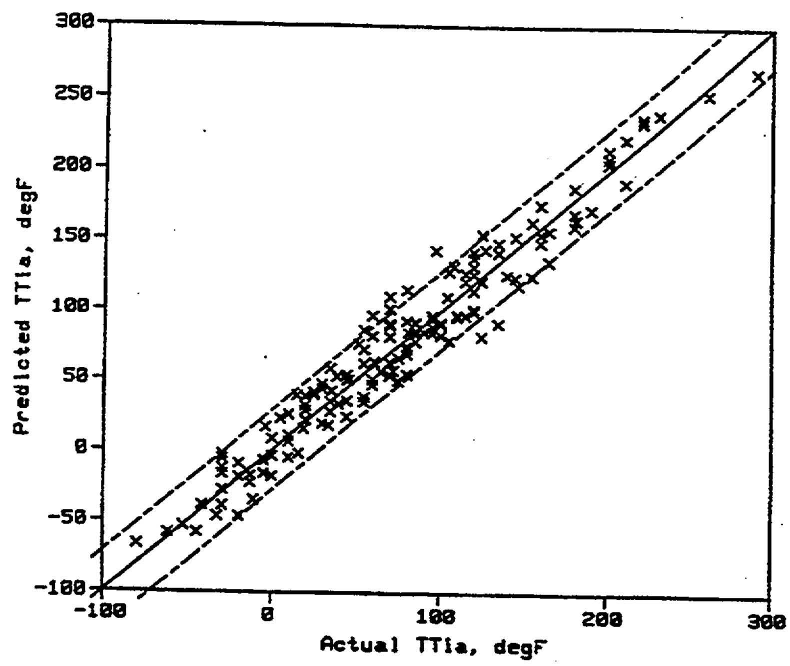


Figure 1. Predicted versus measured values of ΔT for a) the Power Reactor Engineering Data Base and b) the PIA data base.

Effect of Post-Weld Heat Treatment

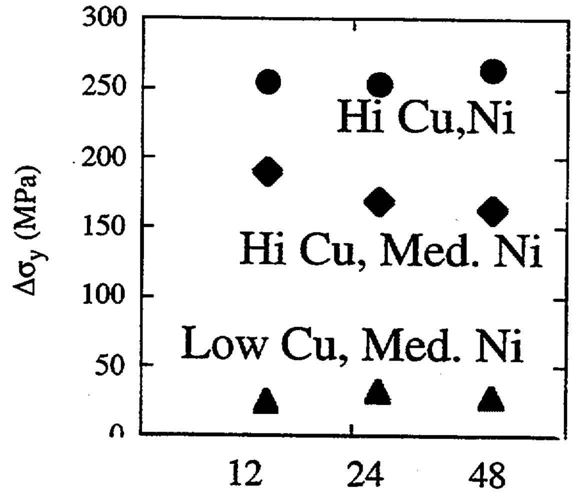
It is now clear that the amount of copper in solution available to precipitate during irradiation and cause hardening/embrittlement is limited by the times and temperatures of the post weld heat treatment (PWHT) [2,6,7]. This is because some of the copper in excess of the solubility limit at the PWHT temperature is removed from solution during the heat treatment, forming relatively large precipitates that do not contribute substantially to hardening. Hence, even though some welds contain bulk copper contents well in excess of 0.4%, the effective maximum copper content, Cu_{max} (that available to precipitate during irradiation) is much less; this has a major effect on reducing the predicted shift for steels with Cu >0.3%. For instance, a revised PREDB fit for Linde 80 welds sets $Cu_{max} = 0.25\%$.

Moreover, preliminary work in our program has indicated that the reduction in radiation hardening from pre-precipitation of copper during PWHT is also a function of the stress relief time, t_{sr}, and Ni content as well as the stress relief temperature T_{sr}. This is illustrated in Figure 2, where radiation hardening after $\phi t = \sim 9 \times 10^{18}$ n/cm^2 at $\phi = 1 \times 10^{12}$ n/cm^2-s, $T_i = 290°C$ is plotted as a function of final stress relief conditions. Figure 2a) shows $\Delta\sigma_y$ as a function of t_{sr} at $T_{sr} = 607°C$ for steels containing various Cu and Ni contents. While there is little effect of t_{sr} on the low copper steel, the hardening decreases with increasing t_{sr} in the high Cu steel containing medium Ni, but increases with t_{sr} in the high Cu steel with high Ni. Figure 2b shows increasing hardening with increasing T_{sr} (less pre-precipitation) and Ni content. Understanding the role of t_{sr} and Ni will be the focus of future microanlytical studies at selected combinations of T_{sr}, t_{sr} and Ni.

Metallurgical Variability

There are significant differences between the leading product form terms in the correlations for the PREDB for welds, plates and forgings which are not understood. These may be the result of non-physical factors and/or untreated variables, such as the Mn content or microstructural differences. Further, through wall gradients and other effects of heat treatment and fabrication details may be significant and particularly beneficial for shallow cracks. Locally inhomogeneous microstructures and chemistries – such as ghostlines, banding, heat affected zones and general fine scale brittle zones – present difficult issues and require estimates of their macroscopic significance. For low Cu steels, where the matrix defects are normally the only embrittling features, there is a large scatter in the data. This is illustrated in Figure 3a, where French surveillance ΔT data [8] for low Cu steels are plotted against ϕt, and compared to the PREDB embrittlement correlation predictions. This large scatter is a reflection of our limited understanding of the nature of the matrix defects and the metallurgical and environmental variables that mediate their evolution and contribution to hardening and embrittlement. As can be seen in Figure 3a and b, the correlation predicts an increasing, unsaturated ΔT with increasing fluence for this component of hardening and embrittlement. As a consequence of the large scatter, predicted embrittlement at high fluence may be non-conservative. This is particularly significant for high Cu steels where a large matrix defect contribution <u>adds</u> to the copper term.

a)



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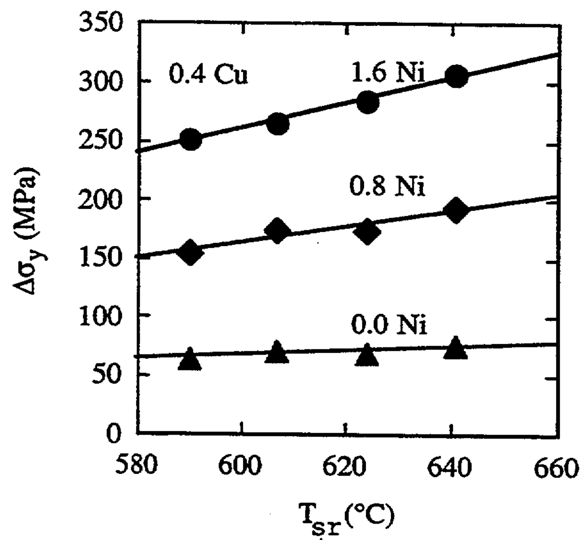
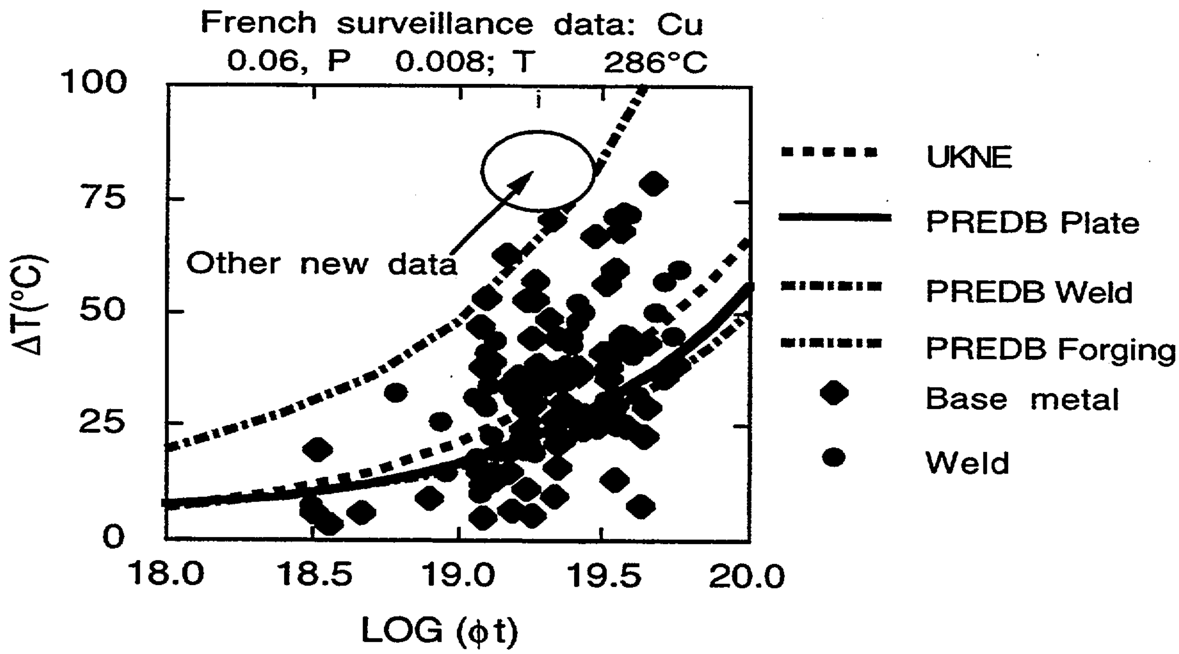


Figure 2. Variation of hardening with a) stress relief time at T_{sr}=607°C and b) stress relief temperature for t_{sr} =48h, for steels with varying Cu and Ni contents

a)



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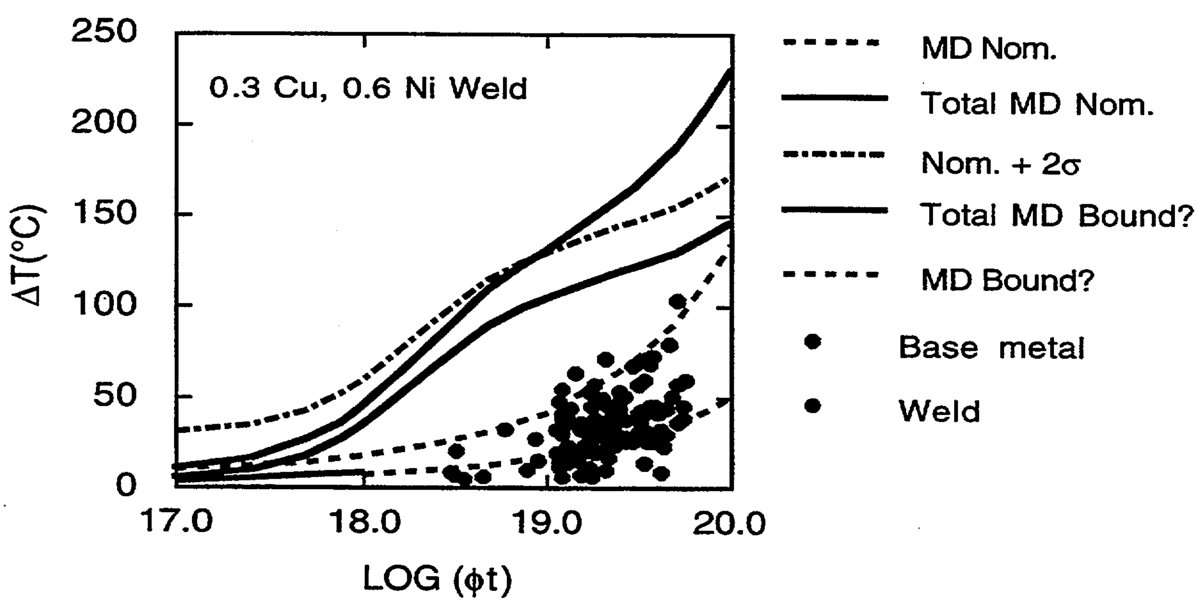


Figure 3 Shift versus fluence: a) comparison of French surveillance data for low copper steels to various shift predictions; b) illustration of possible non-conservatism at high fluence

The dashed-dot line in Figure 3b shows the two standard deviation upper bound on the PREDB prediction. The higher solid line shows the normal copper term plus the upper bound of the matrix defect term. This suggests that above 10^{19} n/cm^2 the nominal PREDB upper bound is not conservative with respect to a larger-higher fluence matrix defect data base. Hence, resolving the nature of these defects and the key embrittlement variables that dictate their evolution is imperative.

Late Blooming Phases

As noted earlier, models and increasing experimental evidence suggest that phases rich in Ni and Mn may form in low Cu steels (e.g., Fig. 4). Results of thermodynamic calculations (Figure 4a) show Mn-Ni rich precipitates are promoted by increasing Ni and Mn content and lower irradiation temperatures. These phases may require a small degree of Cu precipitation to catalyze their nucleation. Hence, they may not contribute to hardening and embrittlement until relatively high fluences. Moreover, because they can be largely constituted of alloying elements which typically occur in concentrations in excess of 2%, they may lead to hardening and embrittlement far in excess of what might be predicted from existing correlations, even for high Cu steels. Such delayed embrittlement could produce a technical surprise that could have serious implications to pressure vessel life extension. Clearly, it is important to understand and quantify the composition-ϕ-ϕt-T_i regime in which they evolve, and develop a better quantitative description of their contribution to embrittlement.

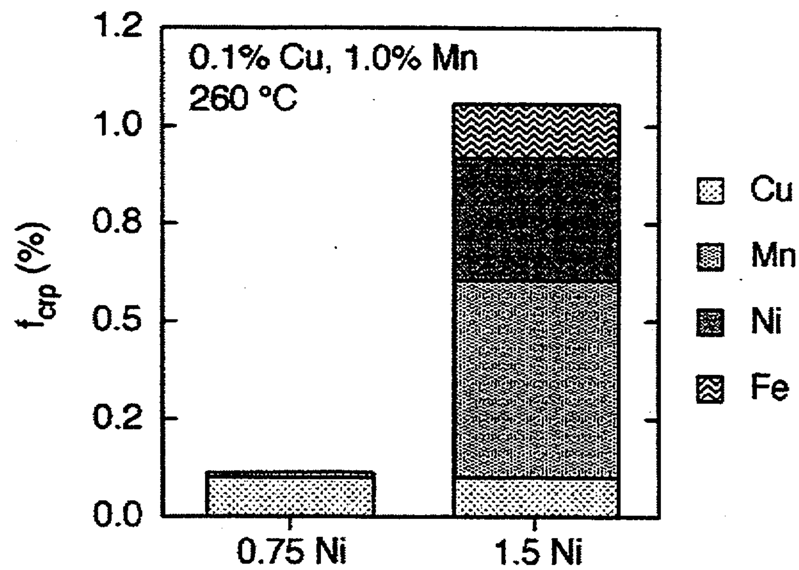
Through Wall Gradients

There is still some controversy over the way in which embrittlement variations through the pressure vessel wall arising from attenuation of the neutron flux should be estimated. Much of the disagreement has focused on what the appropriate exposure unit should be – e.g. whether it should be ϕt>1Mev or displacements per atom, dpa. However, as shown in Figure 5, the choice of exposure unit has a relatively modest effect on the predicted attenuation. In addition, however, the attenuation depends strongly on metallurgical variables such as Cu and Ni, and irradiation variables such as ϕt. This is due to the fact that the exposure dependence, regardless of which unit is chosen, depends strongly on these variables. Figure 5b shows that discrimination of the appropriate exposure unit could also be confounded by through wall variations in the unirradiated transition temperature T_u.

Flux and Time

The NUREG PREDB correlation has a flux-time, ϕ-t, term to account for thermal diffusion in CRP growth kinetics. This is accomplished by adding an increment to the fluence in the Cu/Ni term; the increment is proportional to time and is equivalent to about 1.5×10^{18} n/cm^2 at 10^9s. The consequences are illustrated in Figure 6. The ϕ-t term is significant only in sensitive (high Cu/Ni) steels and only at low ϕ.

a)



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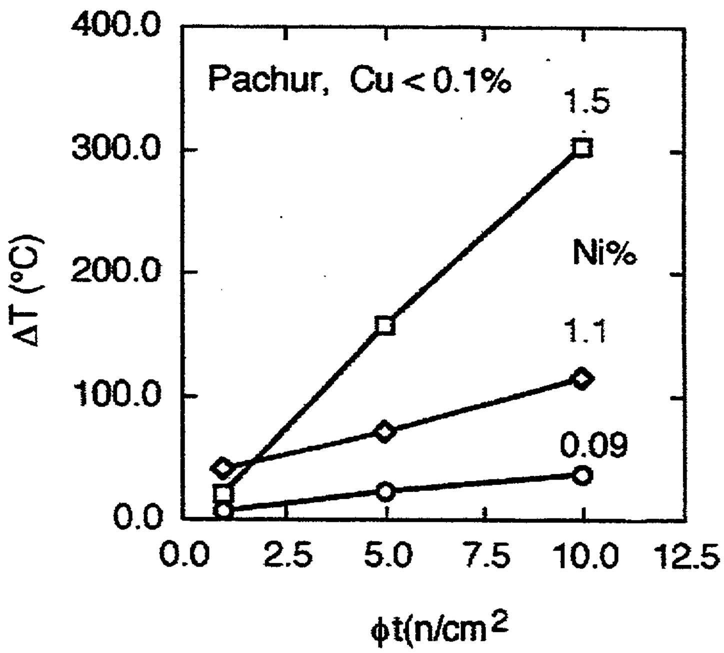
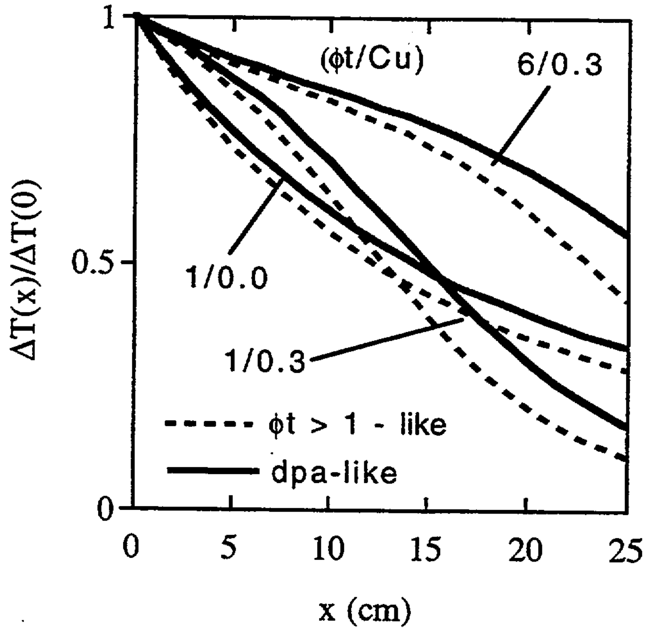


Figure 4 a) Composition of precipitates in irradiated low and high Ni RPV steels, showing a Mn-Ni rich phase in the high Ni steel. b) data from the literature [9] showing evidence of significant embrittlement in low Cu high Ni steels at high fluences.

a)



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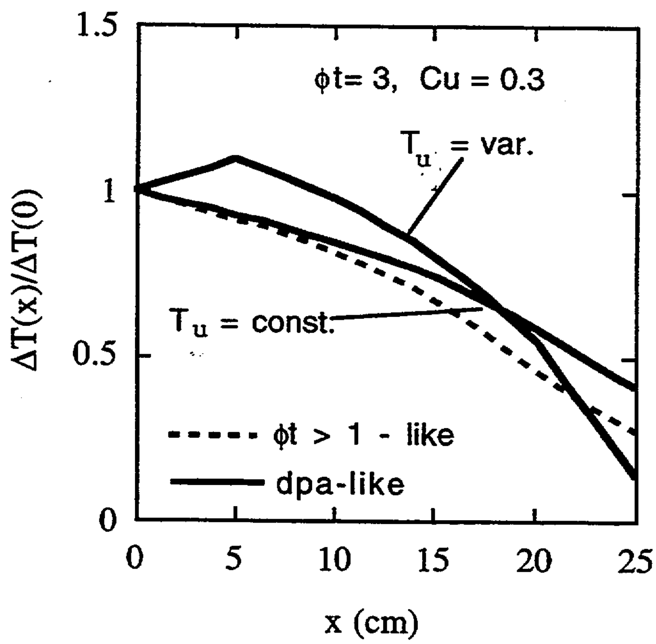
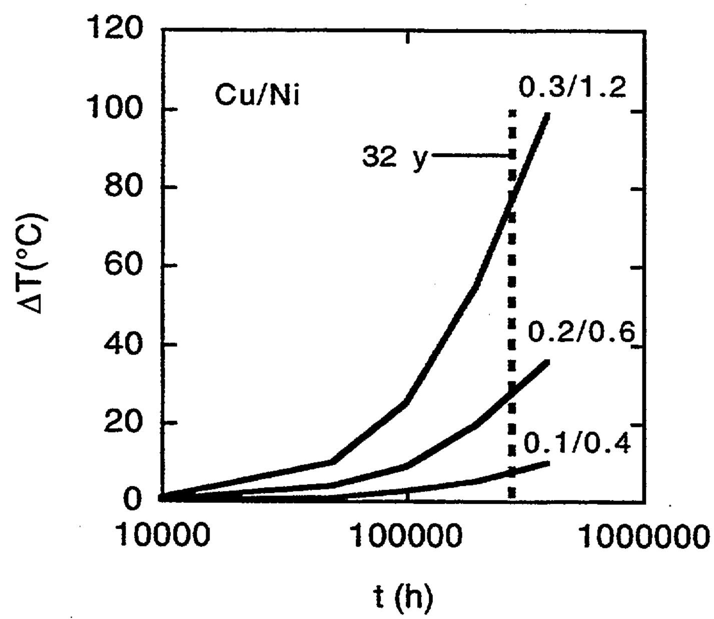


Figure 5 Variation of the shift normalized by the shift at the inside wall with position in the wall thickness: a) for different fluence and copper levels based on two different exposure units; b) for variations in T_u for two different exposure units.

a)



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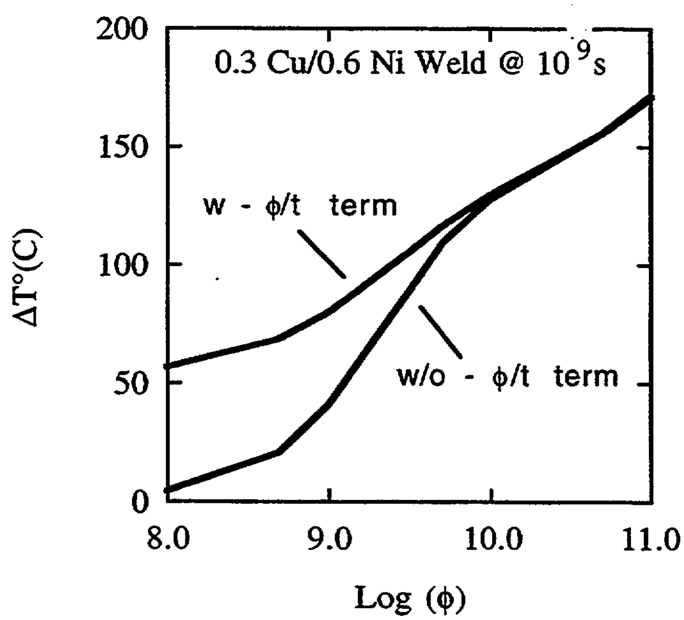

Figure 6 Illustrations of the contribution of the thermal aging term: a) predicted variation of shift with time for RPV steels with various Cu/Ni contents; b) predicted variation of shift with flux with and without the thermal aging term.

Despite this correction, a preliminary re-evaluation of an expanded PREDB by Eason and Wright [10] yielded a statistically significant increment above the baseline ΔT at times exceeding 10^5h. The initial assessment appeared to show no influence of other variables on this increment except irradiation temperature, as shown in Figure 7. This raises the question as to what other mechanisms may be operative at long times. The evaluation is continuing.

THE MASTER CURVE METHOD

The Master Curve (MC) method is a revolutionary, empirically successful approach for measuring transition toughness K_{Jc} as a function of temperature T and for dealing with a number of important issues, including inherent scatter in the data, the rapid increase in K_{Jc} with temperature and size effects.[11,12] The MC method is based on indexing a mean $K_{Jc}(T)$ curve (from which a specified confidence interval can also be determined) to a reference temperature T_o at a reference toughness level. This can be accomplished using a relatively small number of relatively small specimens. The key assumptions include: a fixed shape for $K_{Jc}(T)$, Weibull statistics (with a Weibull modulus of 4), $B^{-1/4}$ size scaling where B is the specimen thickness, a minimum toughness of 20 MPa√m, and a constraint criteria that adjusts data for which K_{Jc} exceeds $(b\ \sigma_y\ E'/30)^{0.5}$, where b is the specimen ligament, σ_y the yield strength, and E' is the plane strain elastic modulus. Despite relative success to date of the MC method, these basic assumptions have not been rigorously tested with single variable type experiments. Hence, work is in progress at UCSB to develop independent verification of the key assumptions, beginning with the B (statistical) and b(constraint) size scaling and the MC shape.

To date, most size effects experiments have varied B and b simultaneously by using specimens of varying size but with self-similar geometries, so that any independent contributions of B or b are masked. To examine the size scaling assumptions, we have begun testing specimens for which either B or b is systematically varied with the other dimension held fixed. Preliminary testing has been done on HSST Plate 02 specimens. A more extensive base set of specimens with dimensional variations in the range 8.25mm $\leq$ B $\leq$ 250mm and 3.05mm $\leq$ b $\leq$ 25.4mm have been cut from a piece of the Shoreham vessel and testing has begun at –91°C. Strong effects of b but not B on toughness have been found in the preliminary testing. However, the initial Shoreham results suggest that $B^{-1/4}$-type scaling is observed – e.g., Figure 8. Completion of this large matrix is necessary to better understand the scaling laws.

The retention of a constant $K_{Jc}(T)$ shape with increasing irradiation has to date been at odds with micromechanical models of cleavage fracture. These models are premised on cleavage initiation occurring when a critical stress σ^* is exceeded over a microstructurally significant dimension (length, area, volume) ahead of a blunting crack. The original models were premised on a single critical event, which was associated with the propagation of a sharp crack from a critically stressed particle (e.g., carbide) of sufficiently large size[13]. However, more recent evidence suggests that cleavage and quasi-cleavage crack initiation occur at a critical accumulation of microcleavage cracks so formed; in such a case, cleavage can be modeled in terms of a critical stressed-area ahead of the crack [5,14]. If σ^* is independent of temperature, these models predict a flattening of the $K_{Jc}(T\ )$ curve with increasing irradiation hardening. Since the peak

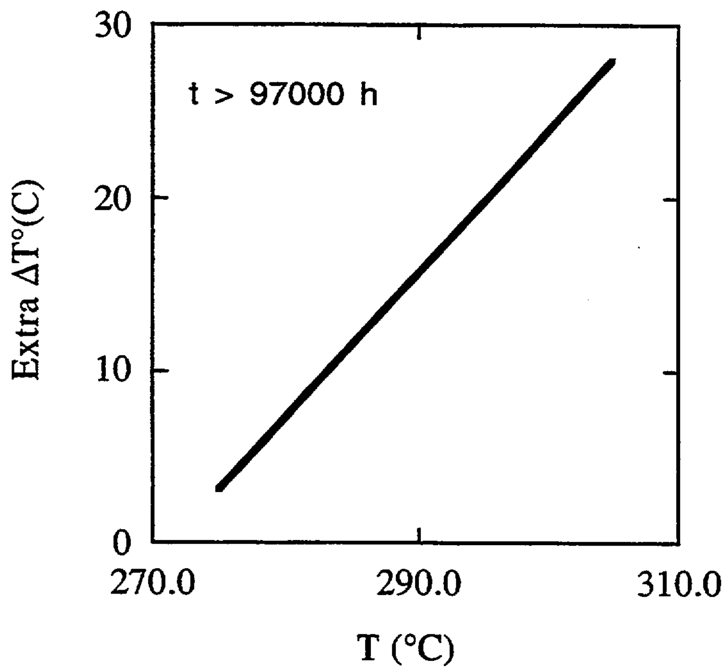

Figure 7 Variation of the increment of shift above the baseline at long exposure times as a function of T_i.

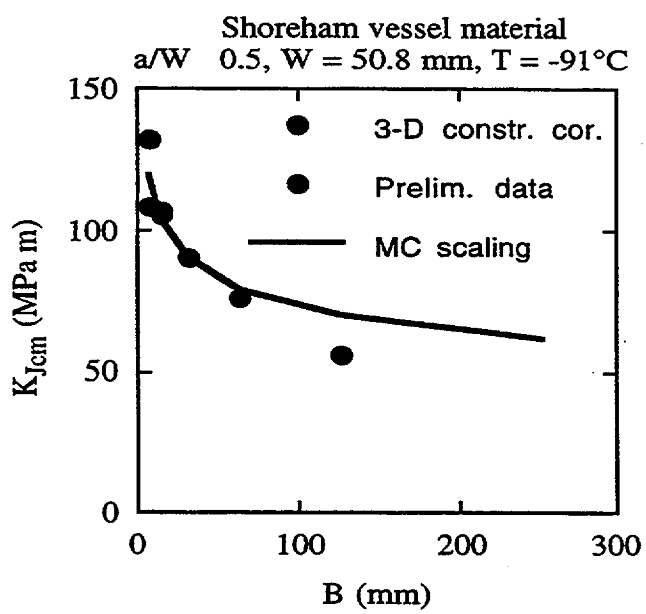

Figure 8 Variation of K_{Jc} with B for a series of preliminary tests on the Shoreham vessel material. The MC scaling is superimposed on the data.

crack tip stresses are proportional to the yield stress, as the material hardens the critical conditions occur at higher temperatures where the temperature-dependence of the yield stress flattens out as shown in Figure 9a. On the other hand, a constant MC shape is consistent with a slight temperature dependence of σ^*, that offsets the yield stress temperature dependence, also shown in Figure 9a. Such dependence of σ^* has been suggested in the literature, including microtoughness of single crystal Fe-Si and W as a function of strain rate compensated temperature, as illustrated in Figure 9b. Using such a temperature dependent σ^* and accounting for increases in the yield stress due to irradiation hardening, the MC shape can be shown to be approximately constant, and the predicted ΔTs are in good agreement with RPV steel data trends [15]. This is illustrated in Figure 10.

Other Issues

There are a number of other issues that are related to the implementation and application of the MC method. How it ties to the Charpy data base and shift correlations and how this data base will be maintained will need to be addressed. Assumptions about the nature of the defects in the vessel for which the MC applies -- including crack lengths, shallow-surface cracks, crack arrest etc. – remain to be resolved. Since the MC method allows direct measures of K_{Jc} from surveillance specimens, this raises the issue of surrogate materials: that is, the degree to which available materials represent the actual vessel. For instance, large Cu and Ni variability can occur within individual welds, and nominally similar weld populations can exhibit variability in ΔT on the order of 125°C. This issue of surrogate materials is not new, but it is a major issue with respect to how to treat margins. Thus, sampling studies of retired vessels provide a tremendous opportunity to test and verify predictive methods. Since sampling studies are limited in their direct general implications, it is absolutely essential that they be conducted within the context of the *overall knowledge base* and used as a test bed to develop improved and integrated methods.

SUMMARY AND CONCLUSIONS

Current methods of shift predictions are generally accurate, but may not fully account for all potentially significant variables and phenomena. The master curve method appears to be an empirically successful approach for measuring transition $K_{Jc}(T)$ data, but the underlying assumptions need to be rigorously verified. In both cases the unresolved issues and opportunities are all amenable to systematic evaluation. The issue of surrogate materials has, and continues to be, a major issue, having impact on the choice of margins. Within proper context, direct vessel sampling studies present a unique opportunity to test and verify predictive methods.

ACKNOWLEDGEMENTS

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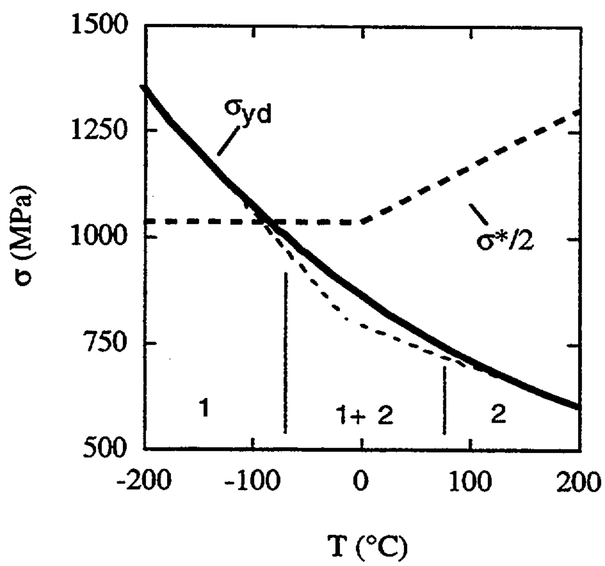
a)



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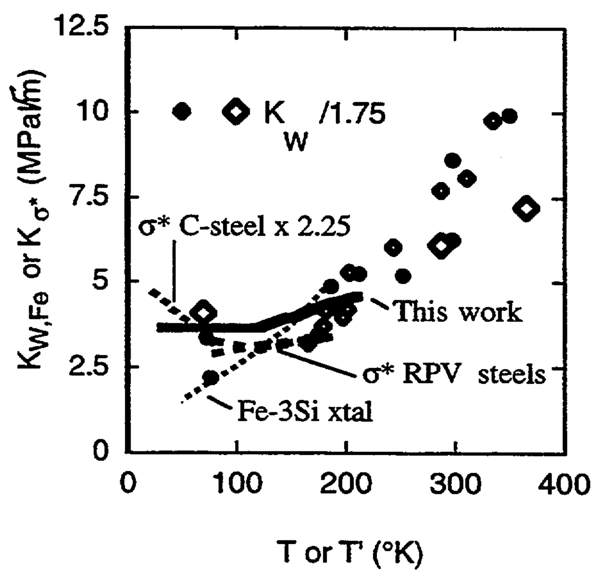
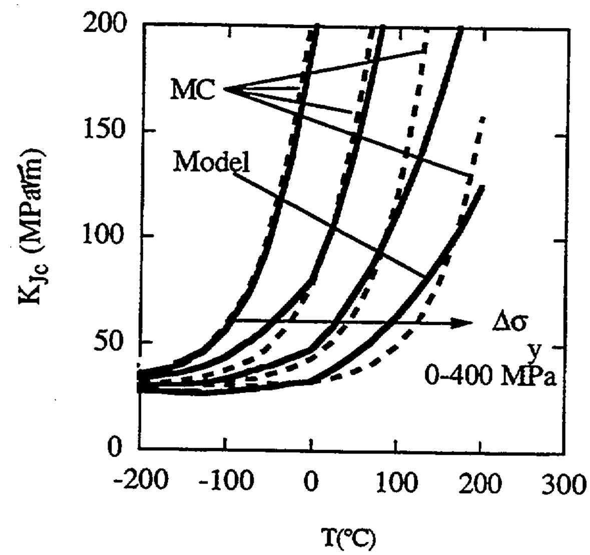


Figure 9 a) Typical temperature variation of the yield stress of RPV steels, and a possible temperature variation of σ* that effect a constant shape MC; b) data suggesting such a temperature dependence of σ* in Fe-Si and W single crystals.

a)



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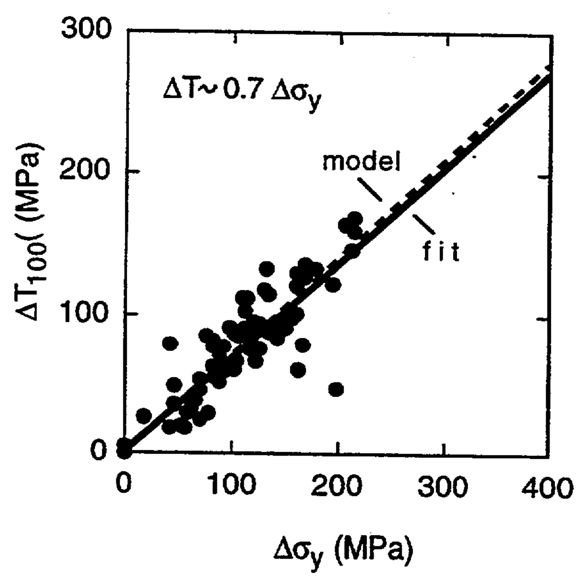

Figure 10 Predictions using a temperature dependent σ^*: a) predicted $K_{Jc}(T)$ curves compared with MC predictions for different levels of irradiation hardening. a) Variation of ΔT with $\Delta\sigma_y$; the predicted variation is the dashed line, while a least squares fit to the data is shown as a solid line

REFERENCES

1. E. Eason, J. Wright, G. R. Odette, “Improved Embrittlement Correlations for Reactor Pressure Vessel Steels,” NUREG/CR-6551, USNRC, November 1998.

2. G. R. Odette, G. E. Lucas, "Recent Progress in Understanding Reactor Pressure Vessel Steel Embrittlement," *Radiation Effects and Defects in Solids*, 144 (1998) 189-231.

3. E. Eason, J. Wright, E. Nelson, G. R. Odette, E. Mader, “Models for Embrittlement and Annealing of Reactor Pressure Vessel Steels,” NUREG/CR-6327, USNRC, 1995.

4. B. D. Wirth, G. R. Odette, M. A. Sokolov, G. E. Lucas and R. D. Klingensmith, SANS Studies of Copper Rich Precipitate Evolution Following Irradiation, Post Irradiation Annealing and Re-Irradiation in Reactor pressure Vessel Steels, submitted to *J. Nuc. Mat.*

5. G. R. Odette and G. E. Lucas, "An Integrated Approach to Evaluating the Fracture Toughness of Irradiated Nuclear Reactor Pressure Vessels," *J. Non-Destructive Eval.,* 14, 3/4 (1996) 137-150.

6. G. R. Odette, G. E. Lucas, B. D. Wirth, C.-L. Liu"Current Understanding of the Effects of Environmental and Irradiation Variables on RPV Embrittlement", Proceedings of the 24th Water Reactor Safety Meeting, October 1996.

7. W. J. Phythian, C. English, J. Nucl. Mater., 205 (1993) 162.

8. C. Brillaud and F. Hedin, In-Service Evaluation of French Pressurized Water Reactor Vessel Steel, Effects of Irradiation on Materials 15, ASTM- STP 1125, R. Stoller, A. Kumar, D. Gelles (eds), American Society for Testing and Materials (1992) 23.

9. D. Pachur, G. Seivers, Irradiation Program for Pressure Vessel Steels, KFA Julich, 1974.

10. E. Eason, J. Wright, G. R. Odette, “Surveillance Data Base and Embrittlement Correlation,” Proceedings of the Research Assistance Task Force on: Radiation Materials Science Applied to Reactor Pressure Vessel Embrittlement, L.K. Mansur (ed), USDOE (1999) 3-18.

11. J. G. Merkle, K. Wallin, D. E. McCabe, “Technical Basis for an ASTM Standard on Determining the Reference Temperature T_o for Ferritic Steel in the Transition Range,” NUREG/CR-5504, USNRC, 1998.

12. ASTM1997-21, Standard Test Method for Determining the Reference Temperature T_o for Ferritic Steel in the Transition Range, American Society for Testing and Materials, 1998.

13. R. O. Ritchie, J. F. Knott, J. R. Rice, "On the Relationship Between Critical Tensile Stress and Fracture Toughness in Mild Steel," *J. Mech. Phys. Solids*, 21 (1973) 395-410.

14. Wirth, B., Edsinger, K., Odette, G. R., Lucas, G. E., "Evaluation of Embrittlement in Pressure Vessel Steels by Fracture Reconstruction," Proceedings of the 17th Symposium on the Effects of Irradiation on Materials, ASTM-STP-1270, American Society for Testing and Materials, Philadelphia, PA (1996) 706-720.

15. G. R. Odette, "A Unified Model for Cleavage Toughness in the Transition," submitted to *J. Nucl. Mater*.

OVERVIEW OF IRRADIATION EFFECTS ON FRACTURE TOUGHNESS AND CRACK-ARREST TOUGHNESS OF RPV STEELS*

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The safety of commercial light-water reactors (LWRs) is highly dependent on the structural integrity of the reactor pressure vessel (RPV). The degrading effects of neutron irradiation on carbon and low-alloy pressure vessel steels have been recognized and investigated since the early 1950s. In those steels at LWR operating temperatures (~288 °C), radiation damage is produced when neutrons of sufficient energy displace atoms from their lattice sites. The defects formed in the steel as a result of those displacements typically cause hardening and a decrease in toughness. Tensile behavior exhibits an increase in yield strength, a decrease in the ultimate to yield strength ratio, and a loss of ductility as measured by specimen elongation. The decrease in toughness is most commonly represented by an increase in the ductile-brittle transition temperature and a decrease of the upper-shelf energy as measured by the Charpy V-notch (CVN) impact test. The synergistic effects of neutron fluence, flux, and spectrum, the irradiation temperature, and the chemical composition and microstructure of the steel must be understood to allow for reductions in uncertainties associated with the development of predictive models of embrittlement. The CVN toughness, however, is a qualitative measure which must be correlated with the fracture toughness and crack-arrest toughness properties, K_{Ic} and K_{Ia}, necessary for structural integrity evaluations.

During the 1960s, it was well recognized that the effects of irradiation could degrade the materials, but definitive effects on fracture properties, especially in thick sections, were not available. Then, the field of fracture mechanics was in the early stages and even the amount of data on other material properties under LWR conditions was deficient. Some experiments showed that the irradiation-induced temperature shift for K_{Ic} and K_{Ia} were approximately the same as that for the CVN impact toughness shift. However, the maximum value of K_{Ic} achieved was only about 66 MPa√m (60 ksi√in.). When the Heavy-Section Steel Technology (HSST) Program was initiated in 1967, the U.S. Atomic Energy Commission had, in fact, already sponsored two irradiation effects projects, and the HSST Program assumed managerial responsibility for them and for the formulation of plans for extensions of those projects [1]. The results from those early programs were important in that they showed irradiation-induced degradation of fracture toughness, a strong temperature dependence of postirradiation fracture toughness, a need for larger specimens, and that the K_{Ic} temperature shift was about the same as the CVN 41-J shift. It is noted, however, that the American Society for Testing and Materials (ASTM) Standard for determination of K_{Ic} was in early development at that time and was not published until 1970. At about the same time, in late 1968, Potapovs and Hawthorne [2] had reported that some residual elements, particularly copper, increased irradiation sensitivity. Subsequently, Title 10, Part 50 of the *Code of Federal Regulations* (10CFR50) evolved to include requirements for fracture toughness of RPVs. Those requirements included surveillance testing with CVN specimens and required fracture toughness specimens if the surveillance materials were predicted to exhibit marginal properties. Furthermore, 10CFR50 requires prediction of radiation effects using the U.S. Nuclear Regulatory Commission (NRC) *Regulatory Guide 1.99* (Rev. 2). Additionally, screening criteria are specified for toughness transition temperatures which, if attained

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by the surveillance tests or by prediction, require plant-specific analyses to demonstrate adequate protection against pressurized thermal shock (PTS). As part of those requirements, 10CFR50 refers to the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code,* Sects. III and XI, for fracture toughness and ASTM E 185 for surveillance testing and analysis as well as application of the test results. The K_{Ic} and K_{Ia} curves of Section XI are shown in Fig. 1; both of these curves were constructed as lower bounds to the existing unirradiated data and were incorporated into the ASME Code in 1972 and 1974, respectively. Moreover, *Regulatory Guide 1.154* incorporates estimates of the variability in fracture toughness and crack-arrest toughness for the critical RPV material in PTS analyses. Thus, it is important to recognize that the developments of fracture mechanics and knowledge of irradiation effects in RPV steels have occurred concurrently; moreover, the rate of developments in fracture mechanics have been significantly compelled by radiation effects research on RPV steels. Although the regulations provide the basic requirements and guidance for testing, they assume: (1) that the CVN temperature shift at the 30 ft-lb (41-J) level is the same as the shift of the reference temperature, $RT_{NDT,}$ (2) that the fracture toughness and crack-arrest toughness transition temperature shifts are the same as the CVN shift, and (3) that the shapes of the fracture toughness curves do not change with irradiation.

In 1972, the HSST Program [1] began a series of irradiation experiments in response to the need for information regarding effects of neutron irradiation on the mechanical properties, particularly fracture toughness, of RPV steels. [In 1989, the HSST Program irradiation effects task was organized into a separate Heavy-Section Steel Irradiation (HSSI) Program.] The irradiation of large fracture mechanics specimens represents a major factor in the HSST and HSSI irradiation series. In terms of linear elastic fracture mechanics, the K_{Ic} measuring capacity increases with the square root of increasing specimen thickness. Many of the irradiation series irradiated 100-mm-thick compact specimens [4T C(T)], a specimen which weighs about 45 kg (100 lb). That size was determined to be the practical upper limit beyond which neutron fluence variations through the thickness and gamma heating would be excessive. The currently completed major projects of the HSSI Program include irradiation effects on (1) dynamic fracture toughness; (2) and (3) ductile tearing resistance; (4) state-of-the-art welds; (5) and (6) temperature shift and shape of K_{Ic} and K_{Ia} curves; (7) stainless steel cladding; (8) commercial low upper-shelf welds; and (9) thermal annealing. A detailed discussion of those series with appropriate references is available in Ref. [3].

Figure 2 shows a summary of representative results from Series 1 through 4 which demonstrated that: (1) high irradiated fracture toughness was attainable; (2) irradiation of high-copper welds causes significant decreases in Charpy upper-shelf and tearing resistance; and (3) current-practice welds exhibit relatively small changes in fracture toughness upon irradiation. As the developments in fracture mechanics have led from the linear-elastic to the elastic-plastic regimes, the specimen size requirements for measuring fracture toughness have significantly decreased. Such experiments have provided results leading to understanding of specimen size effects and, in fact, have contributed significantly to the development of ASTM standard test methods which allow the use of relatively small specimens (e.g., 0.5T and precracked CVN) to establish the material fracture toughness. The results have also provided important data regarding variability of fracture toughness in RPV steels which allow for statistically based analysis. Figures 3 through 5 show examples of variability in copper content and Charpy impact toughness for a submerged-arc weld from the Midland Unit 1 RPV which was never operated. Welds fabricated with copper-coated wire can exhibit significant through-thickness copper variability which will, of course, have a significant effect on predictions of irradiation embrittlement. The figures also show the large variability in transition temperature and upper-shelf energy which exists for a typical RPV weld, even in the unirradiated condition. Table 1 shows that such variability is comparable to those for a weld fabricated without copper-coated wire and for a heavy-section RPV plate steel. Similarly, Fig. 6 shows that a high variability of fracture toughness also exists in the transition temperature region and that positioning of the ASME K_{Ic} and K_{Ia} curves on the basis of RT_{NDT} can lead to significant overconservatism for some steels in the unirradiated condition.

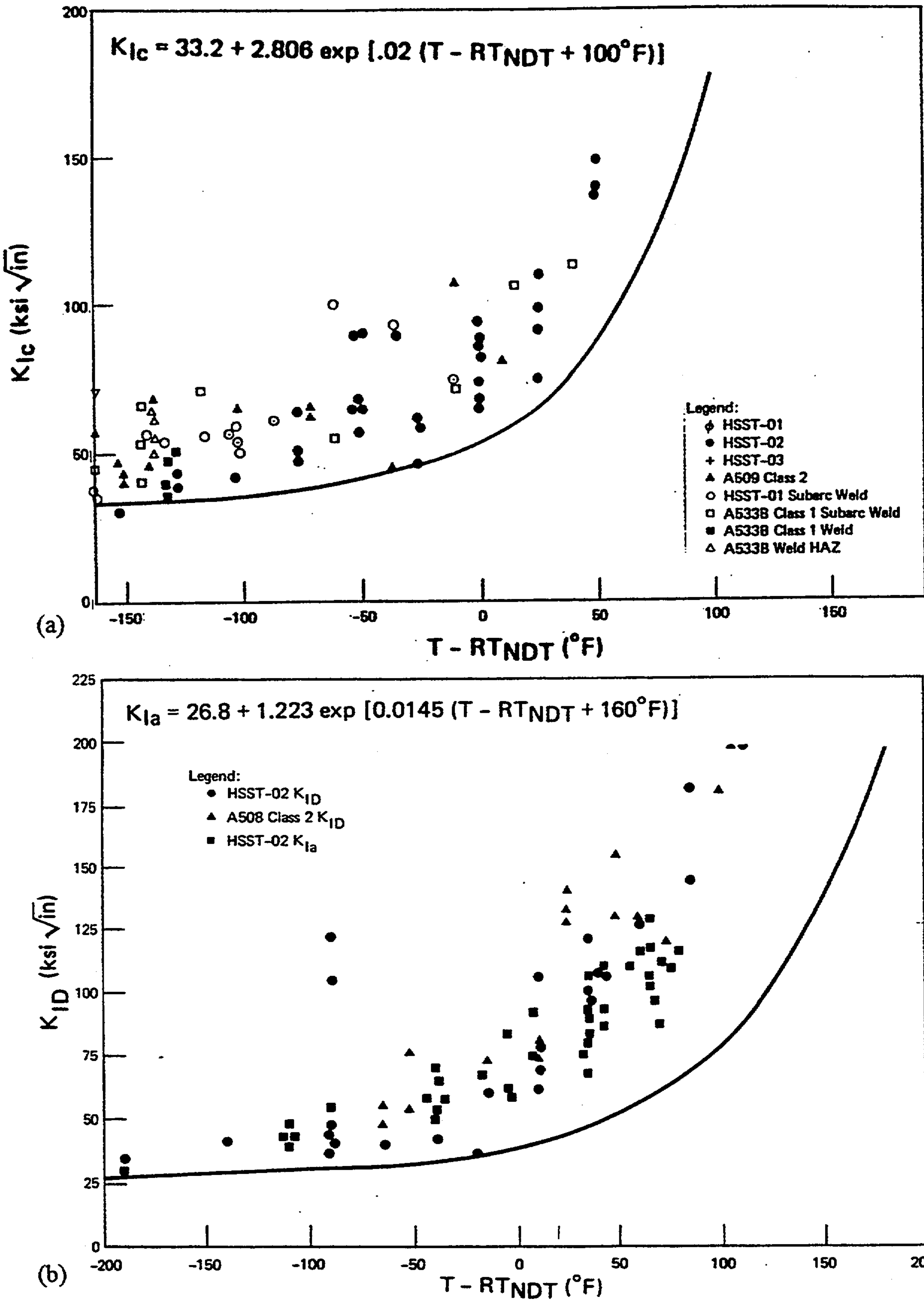


Fig. 1. Plots of the (a) K_{Ic} and (b) K_{Ia} curves in Section XI of the ASME *Boiler and Pressure Vessel Code* and the data used for their construction.

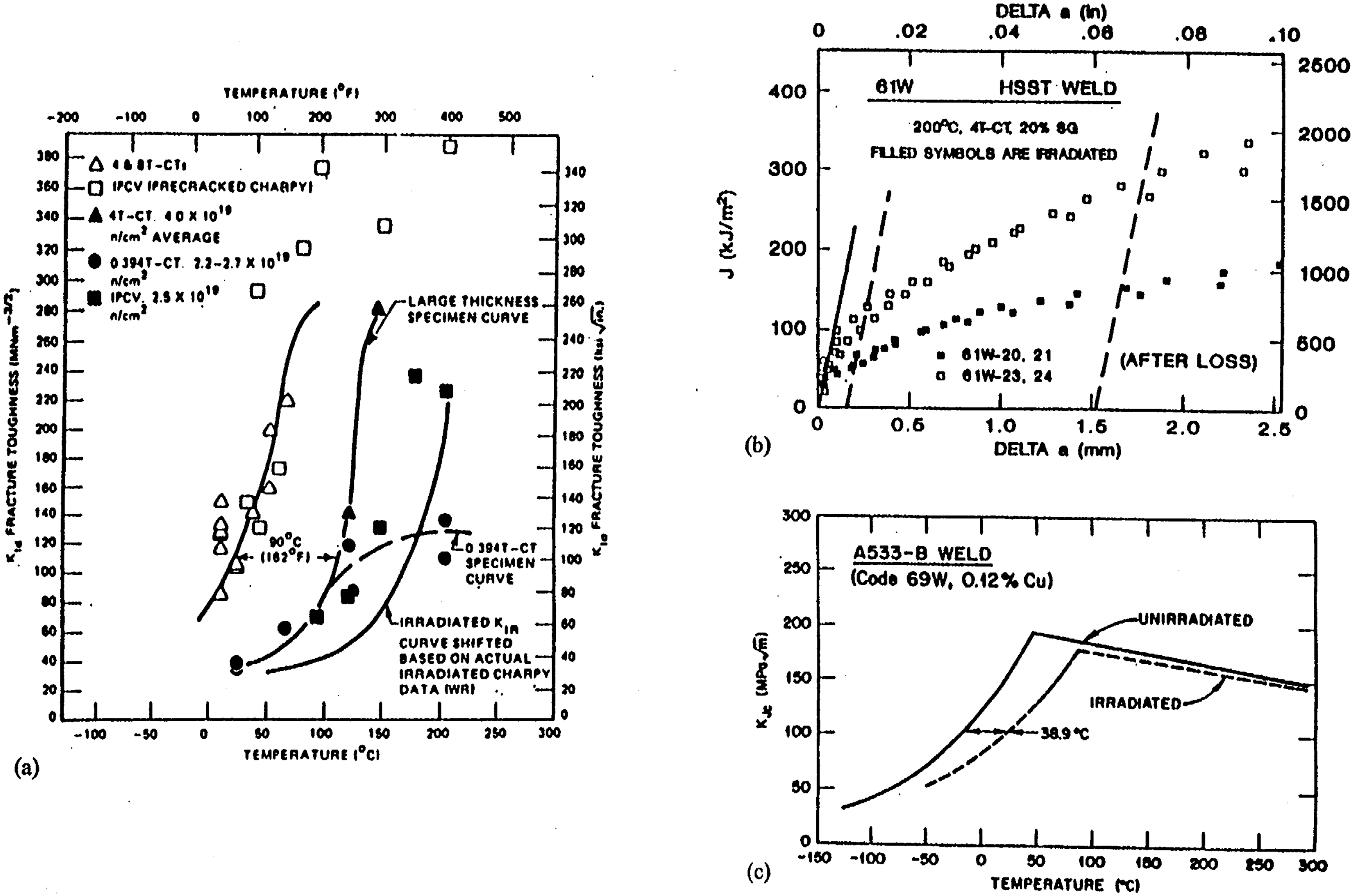


Fig. 2. Plots of representative results from the first four HSST Irradiation Series showing, (a) Series 1 on dynamic fracture, (b) Series 2 and 3 on ductile tearing resistance, and (c) current practice (i.e., low copper and nickel) welds.

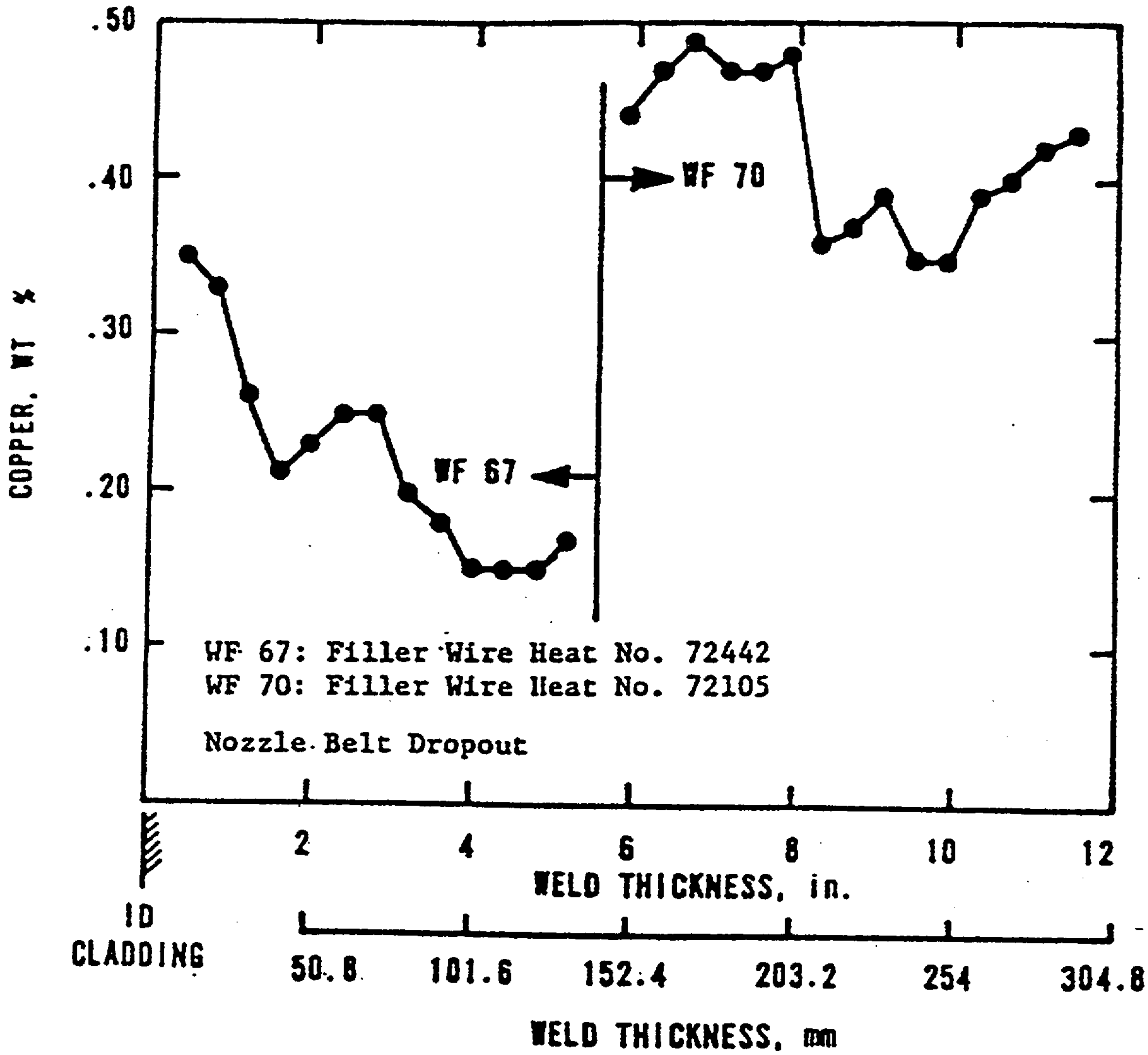


Fig 3. Plot of copper concentration through the thickness of the Midland Unit 1 RPV nozzle course weld WF-70 (from BAW 1599).

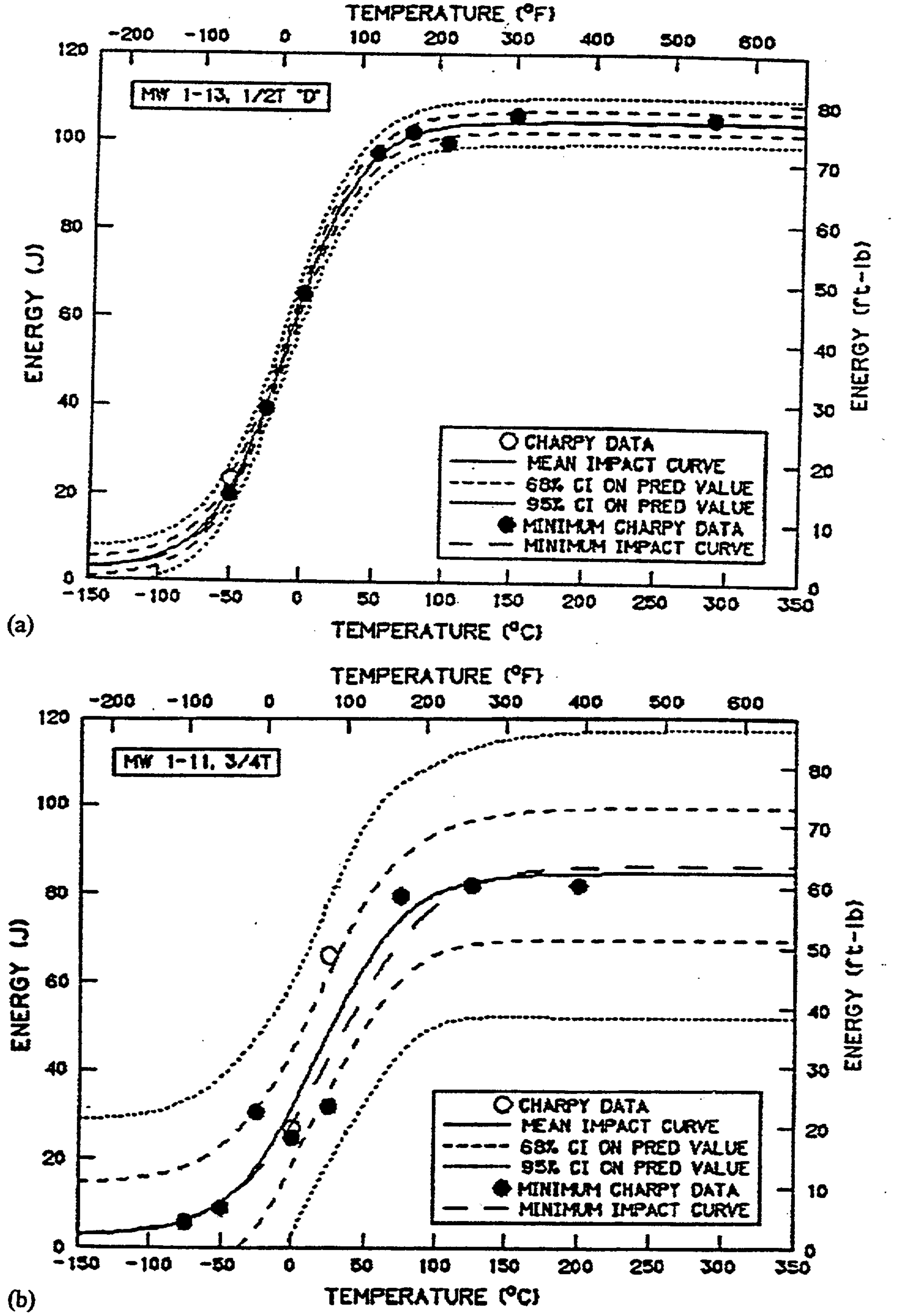
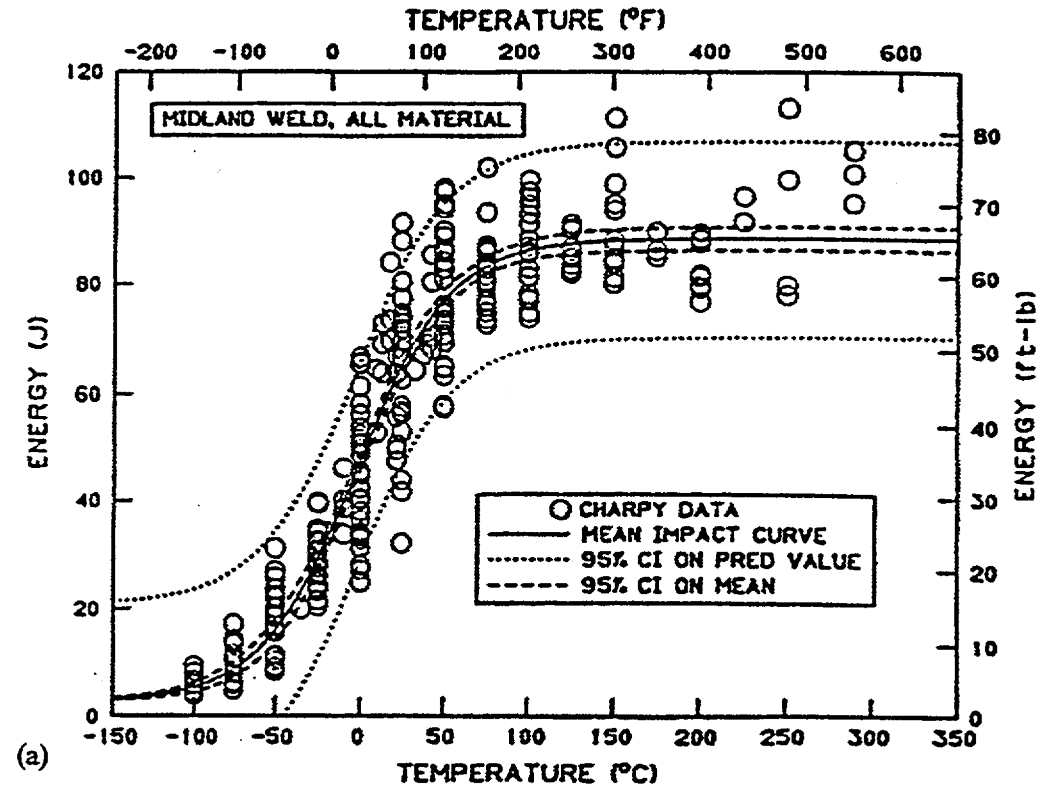
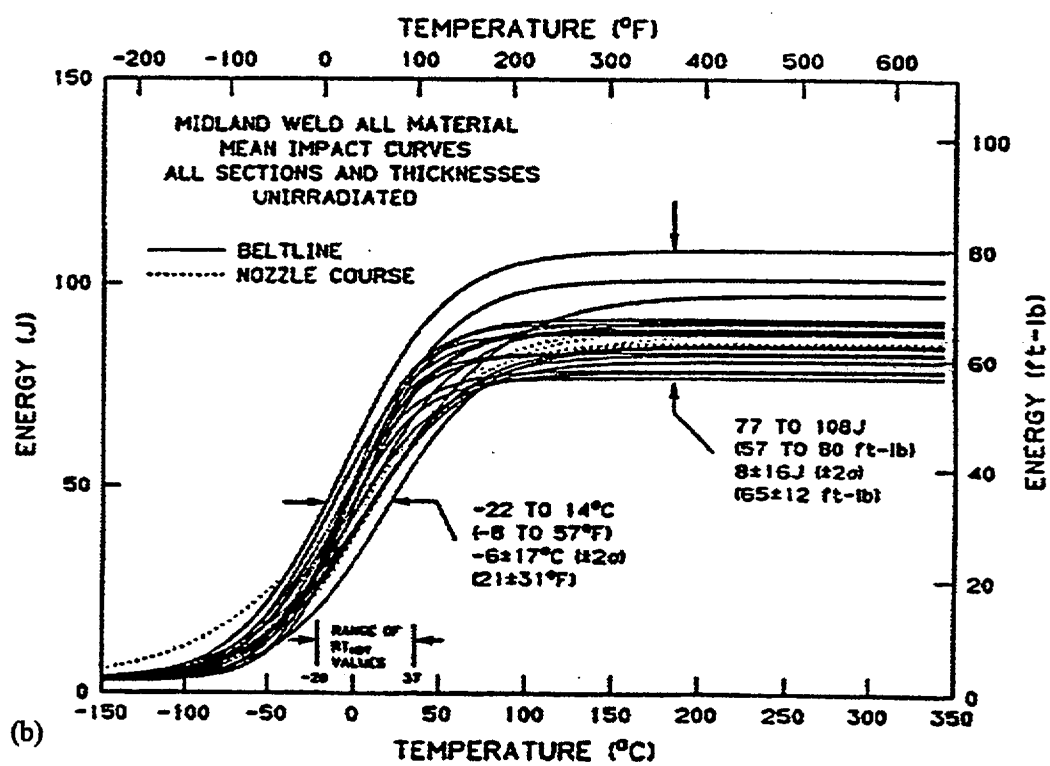


Fig. 4. Plots of Charpy impact energy vs temperature for two different data sets, both from specimens of the Midland Unit 1 RPV beltline weld WF-70, showing 41-J temperatures of (a) -25°C (-12°F) and (b) 14°C (57°F) and with vastly different confidence intervals.



(a)



(b)

Fig. 5. Charpy impact energy vs temperature for Midland Unit 1 RPV weld WF-70 showing (a) a large amount of scatter for the total database with 95% confidence intervals of 53 °C (95 °F) on the predicted value, and (b) for 25 separate Charpy curves the 2σ variations are 17 °C (30 °F) for the 41-J temperature and 16 J (12 ft-lb) for the upper-shelf energy.

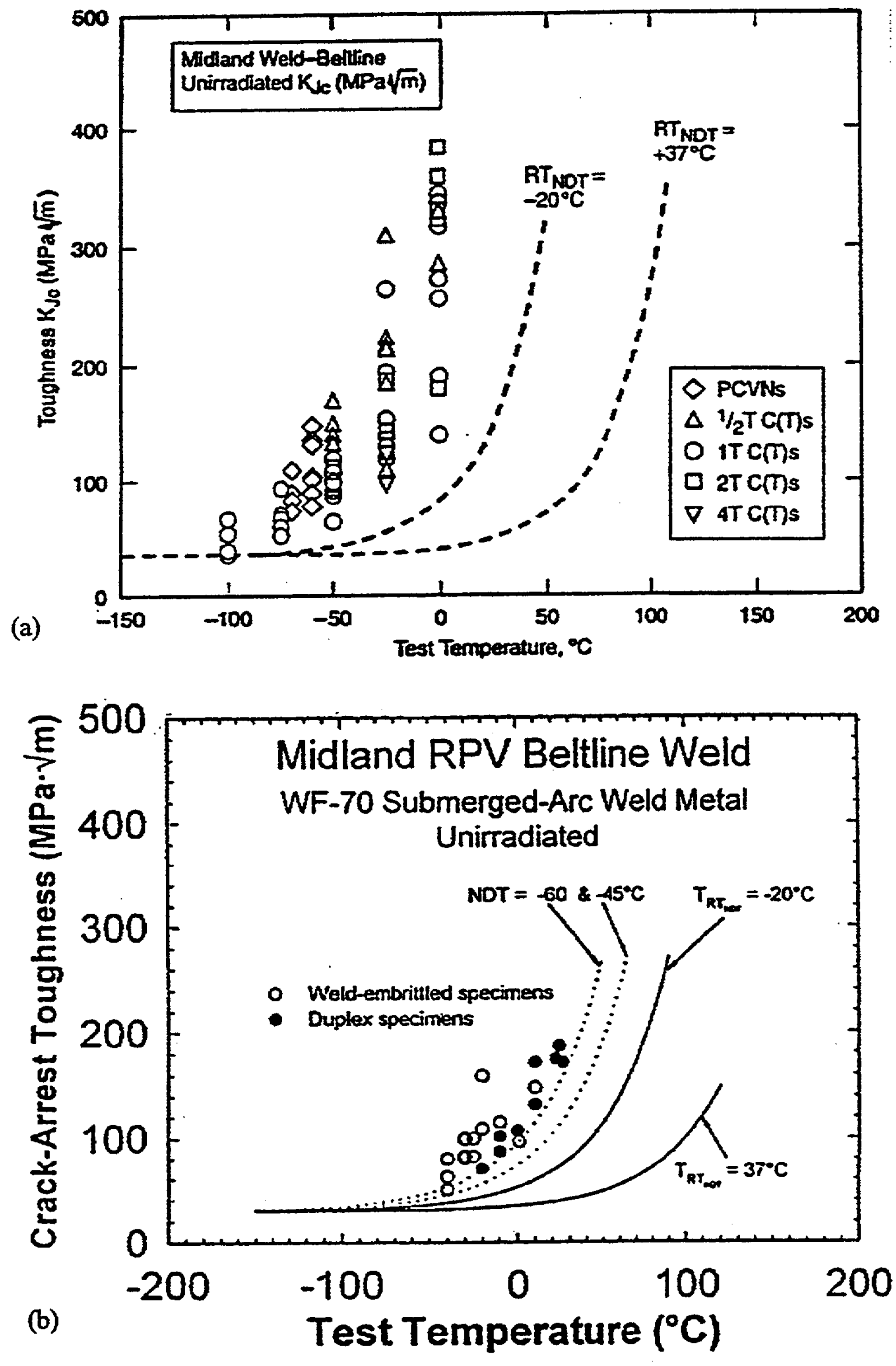


Fig. 6. Plots of (a) fracture toughness, K_{Jc}, and (b) crack-arrest toughness, K_{Ia}, data for Midland Unit 1 RPV beltline weld WF-70 showing the relationships of the data and the ASME curves positioned relative to the highest and lowest values of RT_{NDT} for that weld.

Table 1. Variation (2σ) of the Charpy impact 41-J temperatures for 25 data sets of the Midland Weld compared with those for HSSI Weld 72W and HSST Plate 01 .

Material designation	Material type	Number of CVN curves	2σ on T_{41J}, °C (°F)
Weld 72W	Sub-arc weld with copper in melt, Linde 0124 flux	Monte Carlo simulation of CVN data set with 84 points	13 (23)
Midland RPV weld	Sub-arc weld with copper-coated wire, Linde 80 flux	25 curves, 10-12 points per curve	17 (31)
HSST Plate 01	A533-B-1 plate, 12 in. thick	13 curves, 15-20 points per curve	17 (31)

As mentioned earlier, two key issues are the relationship between the irradiation-induced transition temperature shifts for CVN toughness, fracture toughness, and crack-arrest toughness, and the shape of the fracture toughness curves as a consequence of irradiation (see Fig. 7). Figures 8 and 9 show a comparison of those results for a high-copper (not copper-coated wire) submerged-arc weld for which sufficient testing was performed to allow for meaningful statistical analysis. Similar experiments have provided key results on the effects of irradiation on crack-arrest toughness and have demonstrated a decrease in the temperature difference between the K_{Ic} and K_{Ia} curves with irradiation embrittlement, as demonstrated in Fig. 10. Figure 11 demonstrates that, for two high-copper welds tested in the Fifth Irradiation Series, the fracture toughness shift was greater than the CVN shift and the shape of the fracture toughness curve was slightly changed. Statistical analysis of data from around the world have indicated that the assumption of equivalent irradiation-induced CVN 41-J and fracture toughness transition temperature shifts may not be appropriate for all RPV steels. Figure 12 shows two results of analyses [4,5] of the K_{Ic} database which was used as the basis for the development of the lower-bound K_{Ic} curve in the ASME Code (shown in Fig. 1). The data are analyzed using the "master curve" approach developed by Wallin [6], and show that such a procedure provides appropriate statistical bounds to the database. Figures 13 and 14 show similar analyses [7] of a larger database assembled from the literature for cleavage fracture toughness data, to include elastic-plastic results obtained within the range allowed by the recently developed ASTM E-1921. These two figures demonstrate that: (1) the scatter of fracture toughness for a large number of data is well described by the master curve approach; and (2) that the available fracture toughness data normalized by T_{100} (the transition temperature index used in the master curve approach) do not suggest a change in the shape of the master curve due to irradiation, at least within the range of transition temperature shifts included in the database, about 100°C. In Fig. 15, comparisons of the CVN and K_{Jc} shifts show that on average the shifts are about the same for weld metals but that the fracture toughness shift is greater than the CVN shift by about 16% [7]. Both of these correlations are typified by extremely large uncertainties, with 95% confidence limits up to ±40°C. In a similar fashion, Fig. 16(a) shows that there is a good linear correlation between the T_{100} and CVN 41-J transition temperatures for the steels in the database, but also with a large uncertainty of ±40°C [7]. In Fig. 16(b) the relatively sparse crack-arrest toughness data for irradiated RPV steels indicates that the K_{Ia} shift is about the same as the CVN 41-J shift.*
In this case, there are not enough data to separate base and weld metals.

*S. K. Iskander, unpublished research, Oak Ridge National Laboratory, Oak Ridge, Tenn., 1998.

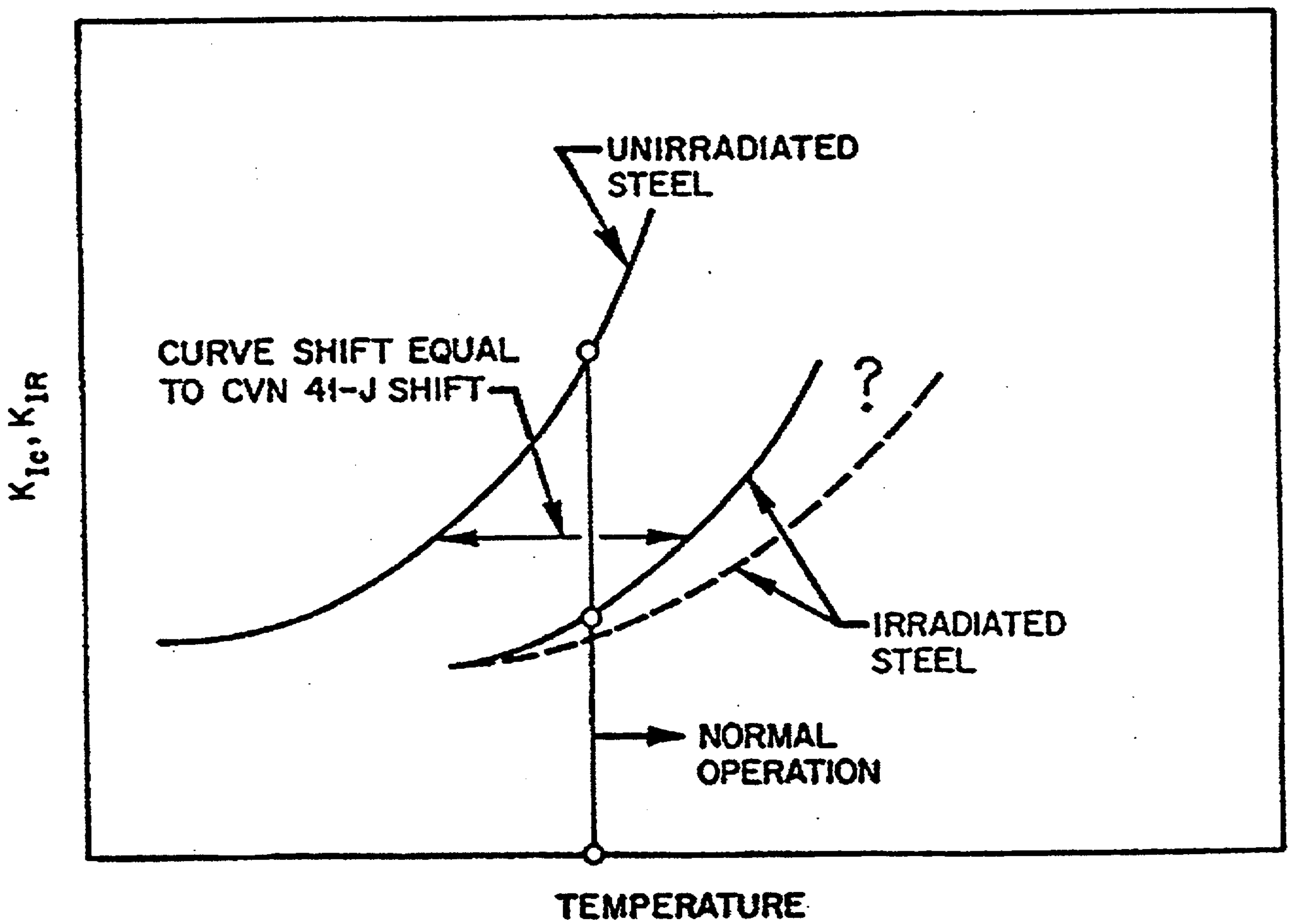


Fig. 7. Schematic drawing depicting a major goal of the HSSI Program to determine the effects of irradiation on the shifts and shapes of the fracture toughness and crack-arrest toughness curves.

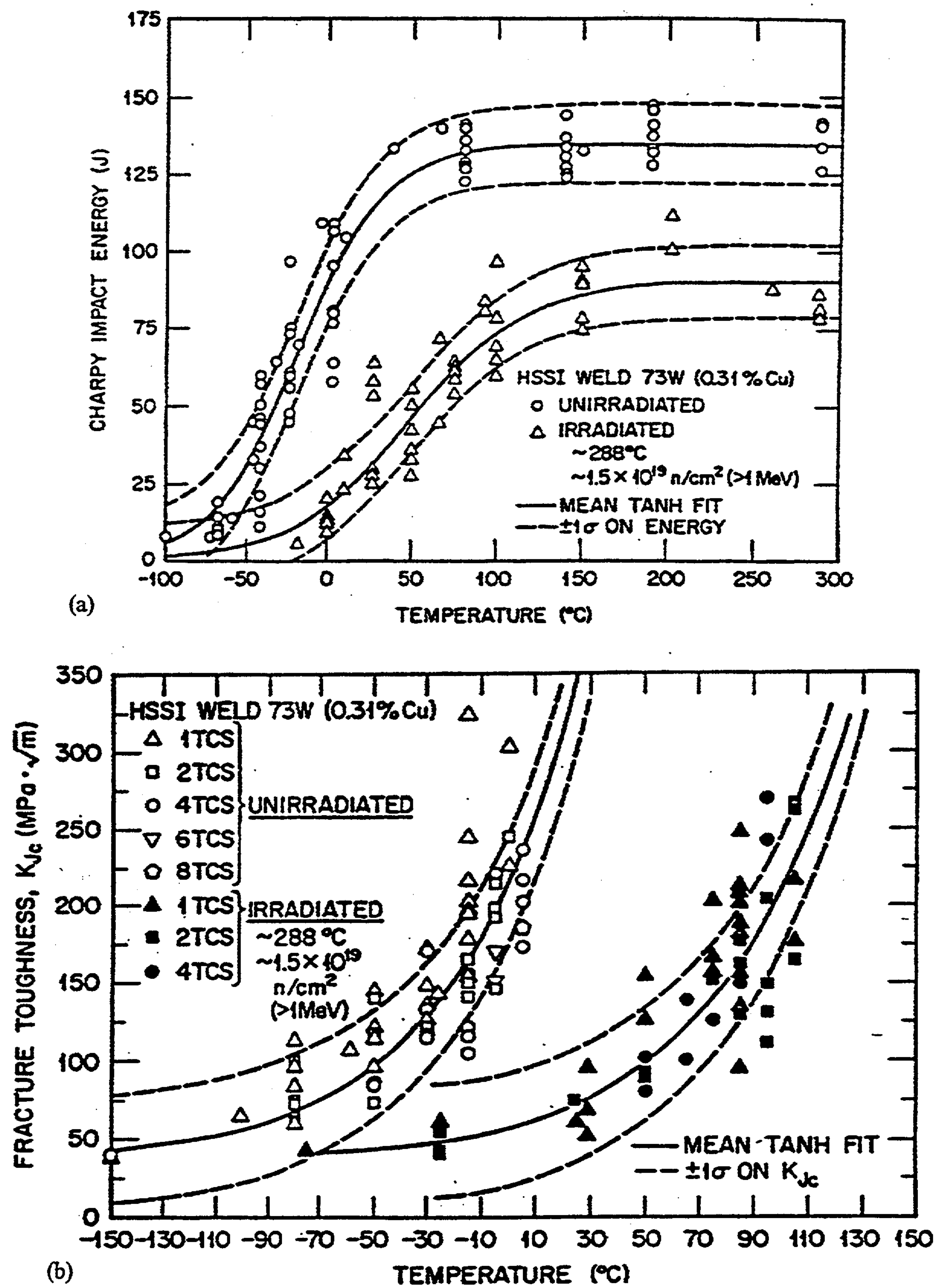


Fig. 8. Plots of (a) Charpy impact energy and (b) fracture toughness vs temperature for unirradiated and irradiated HSSI Weld 73W showing similar scatter in the two measures of toughness.

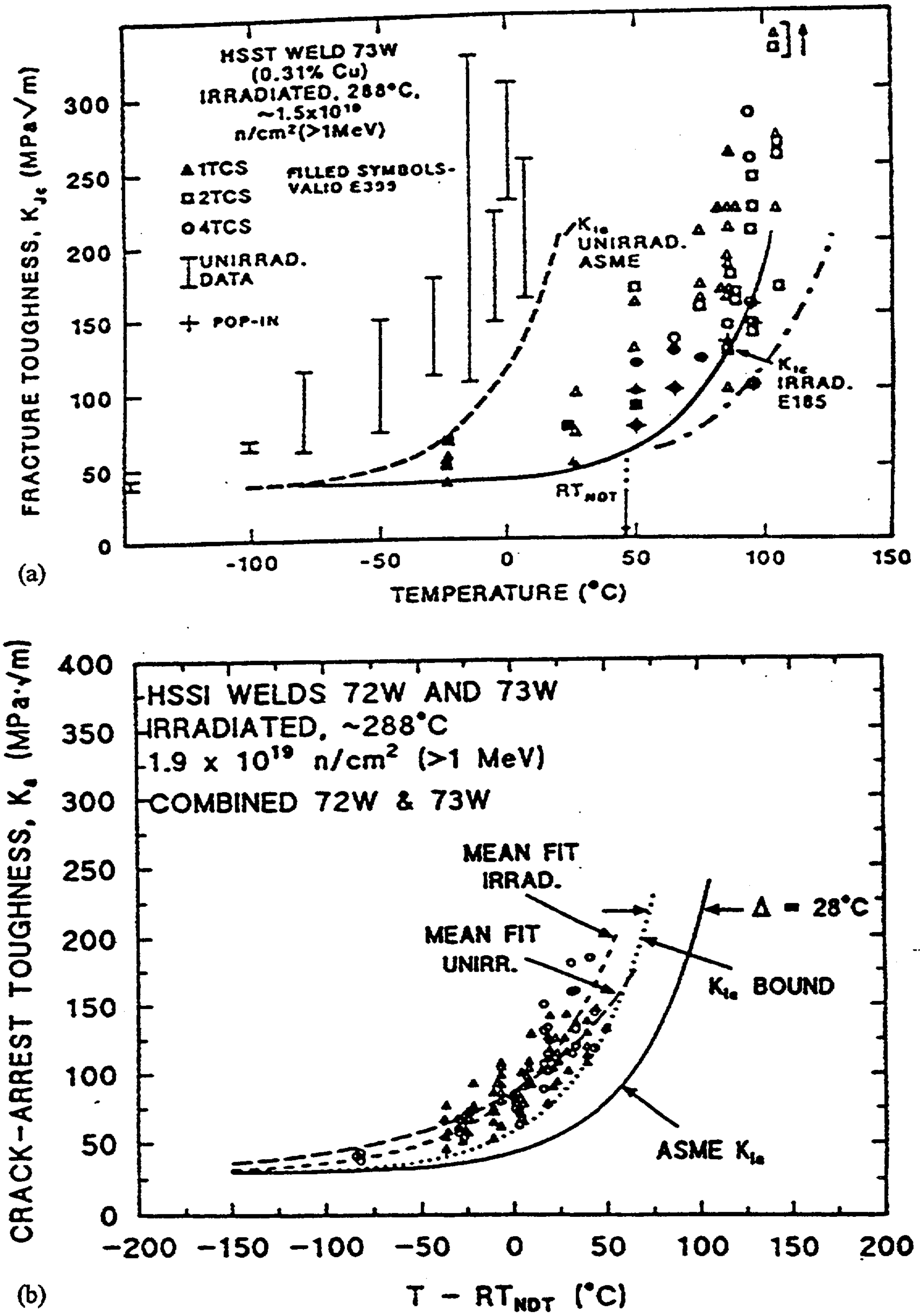
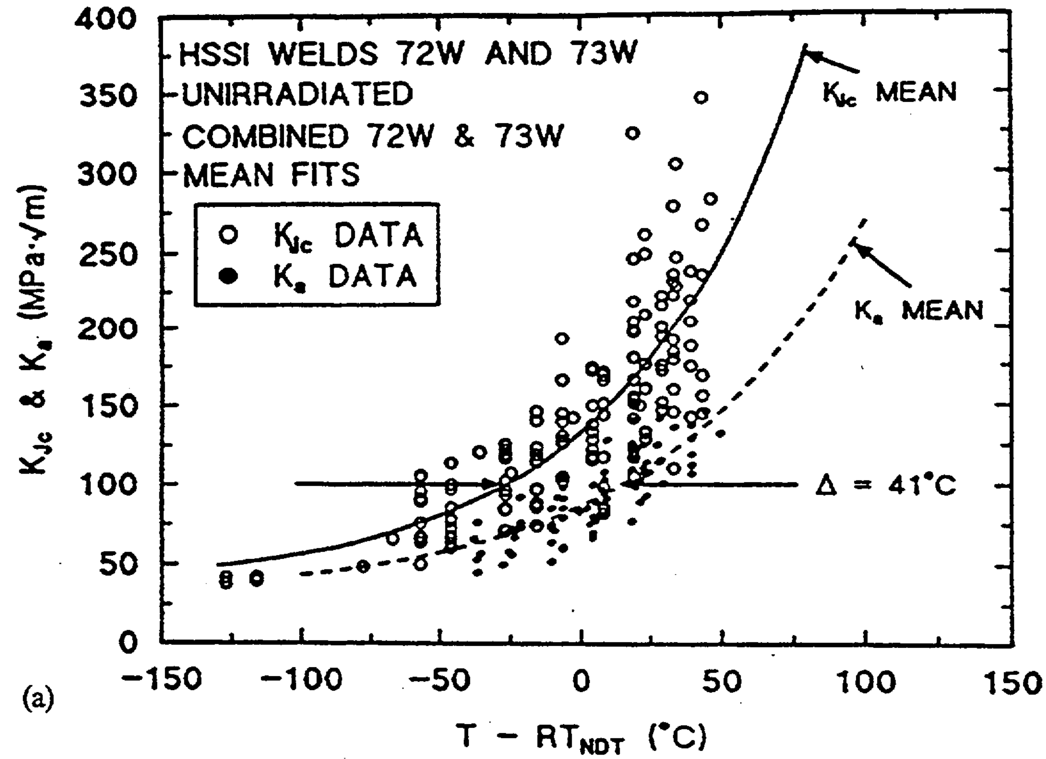


Fig. 9. Plots of (a) fracture toughness and (b) crack-arrest toughness data for unirradiated and irradiated HSSI Weld 72W and 73W. The fracture toughness shift was greater than the Charpy 41-J shift while the crack-arrest toughness shift was about the same as the Charpy shift.

(a)



(b)

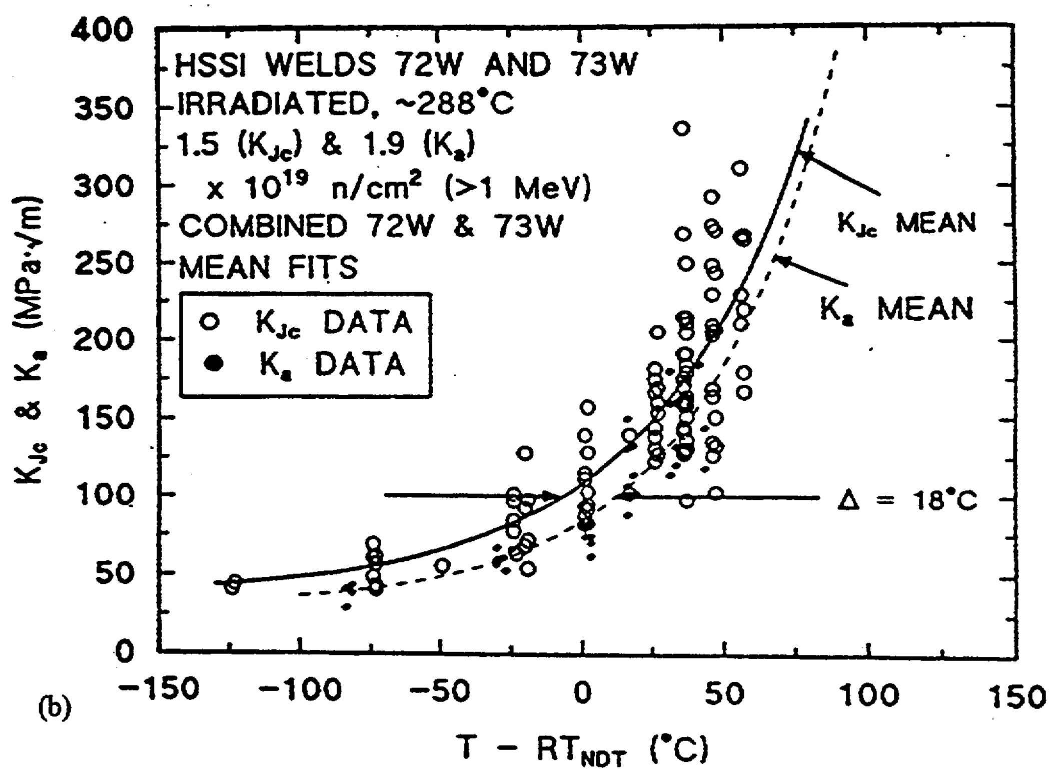


Fig. 10. Plots of fracture toughness and crack-arrest toughness data for (a) unirradiated and (b) irradiated HSSI Weld 72W and 73W vs normalized temperature, showing that irradiation exposure reduced the temperature difference between the mean curves.

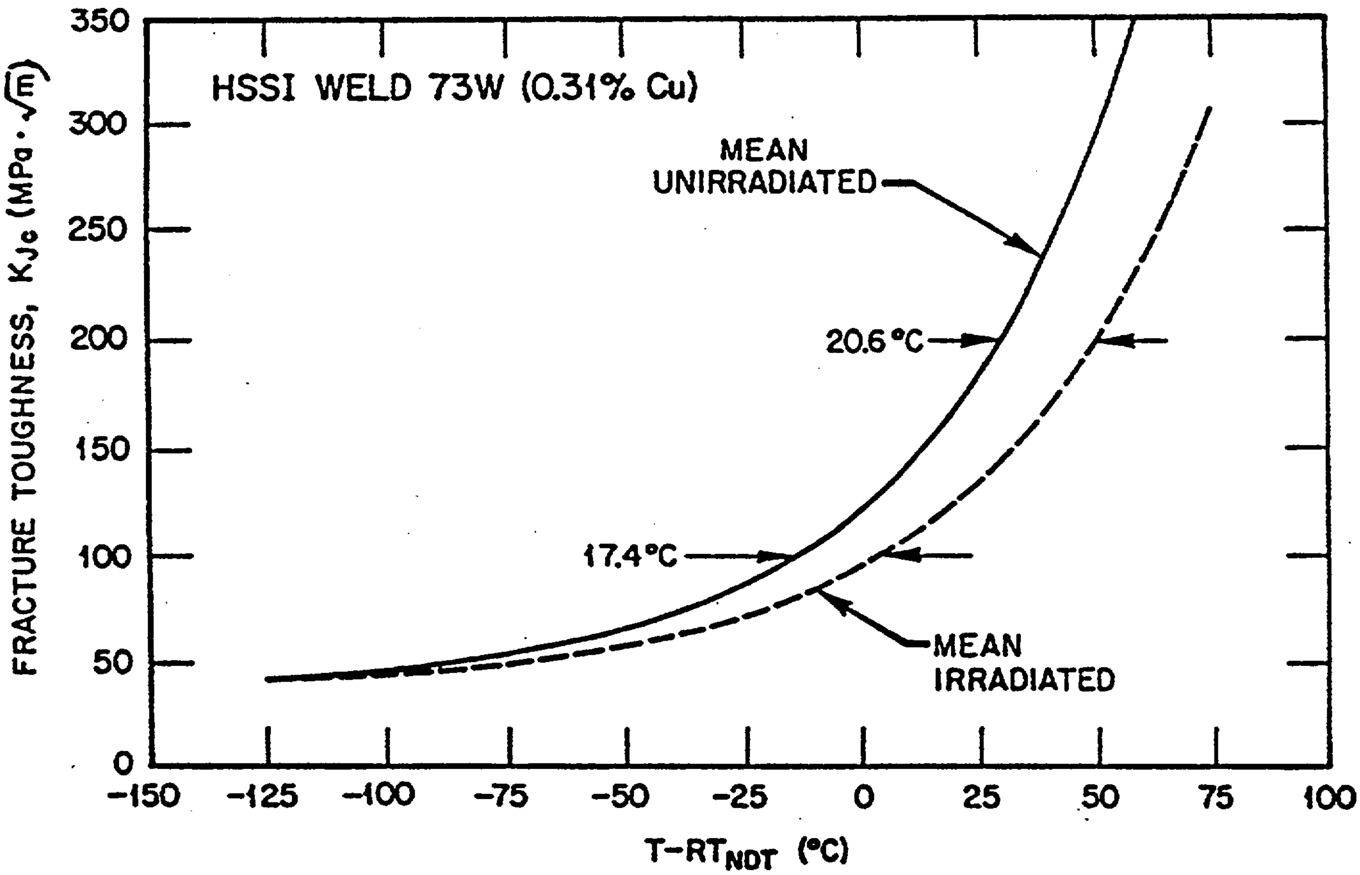
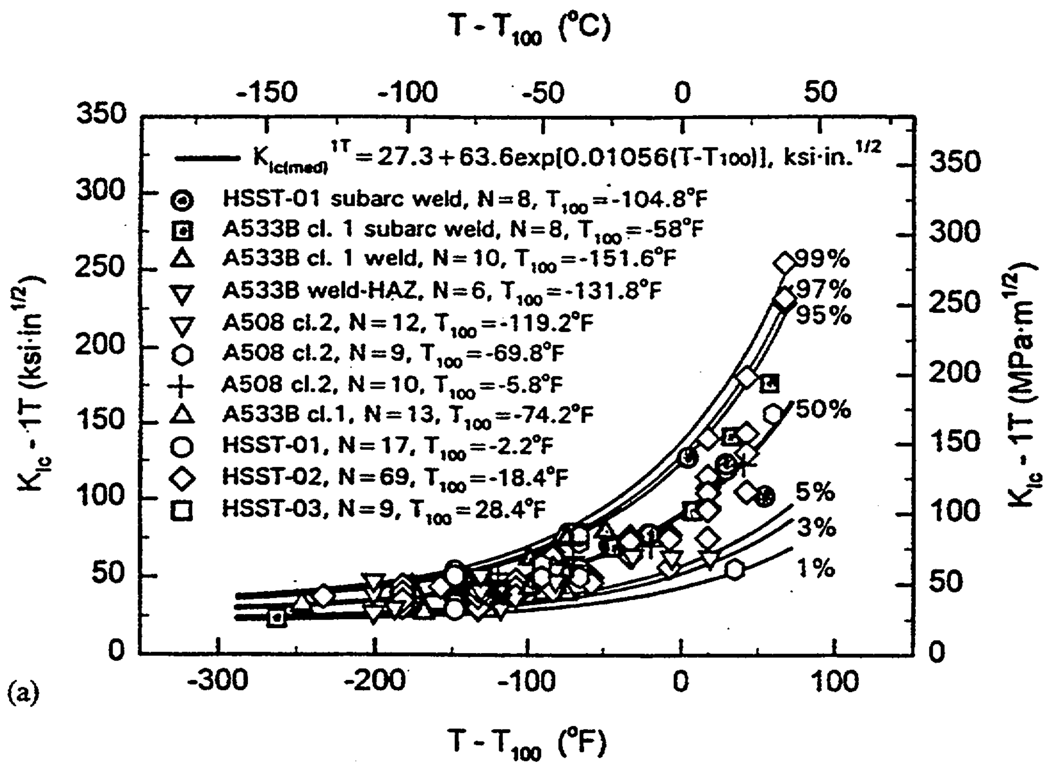
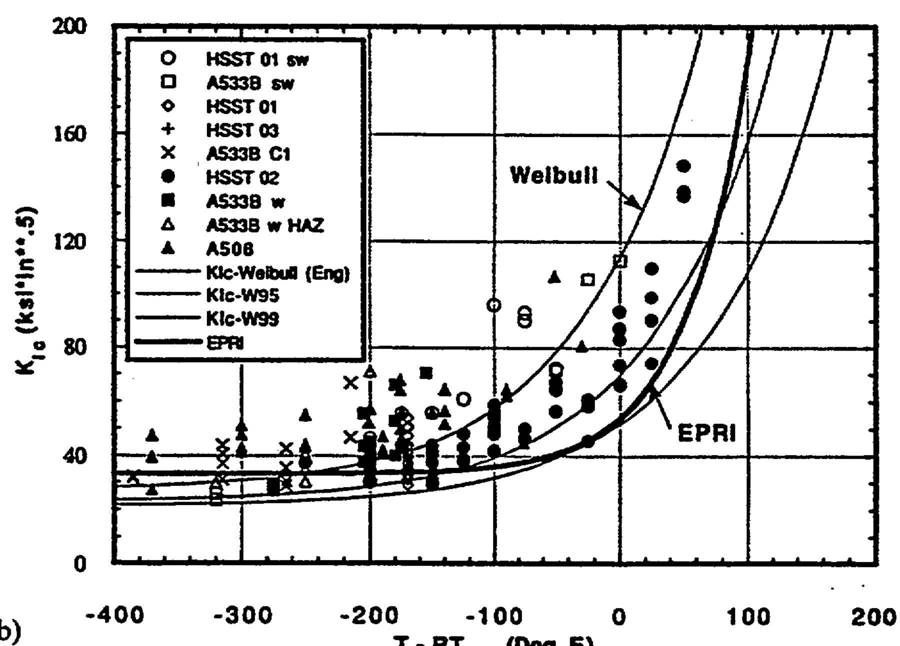


Fig. 11. Mean curves fit to the unirradiated and irradiated fracture toughness data for HSSI Weld 73W vs normalized temperature, showing the fracture toughness shift was greater than the RT_{NDT} shift (equal to the Charpy 41-J shift) and that the slope of the fracture toughness curve was slightly degraded by irradiation.



(a)



(b)

Fig. 12. Plots of the K_{Ic} database used for construction of the curve in the ASME Code analyzed in two different ways; (a) separate master curve analysis for each material in the database and normalized to T_{100}, and (b) regression analysis and master curve analysis of overall data set and normalized to RT_{NDT}.

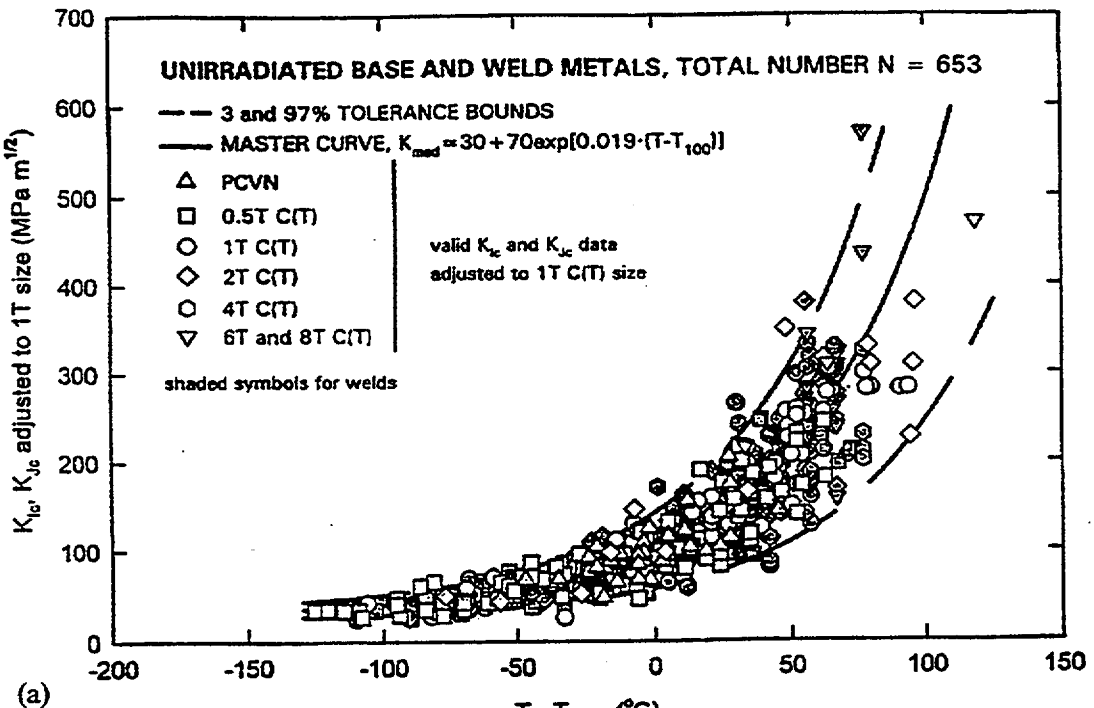


Fig. 13. Plot of fracture toughness data for database of RPV base and weld metals in the unirradiated condition showing that the scatter is large but the data are well described by the master curve concept.

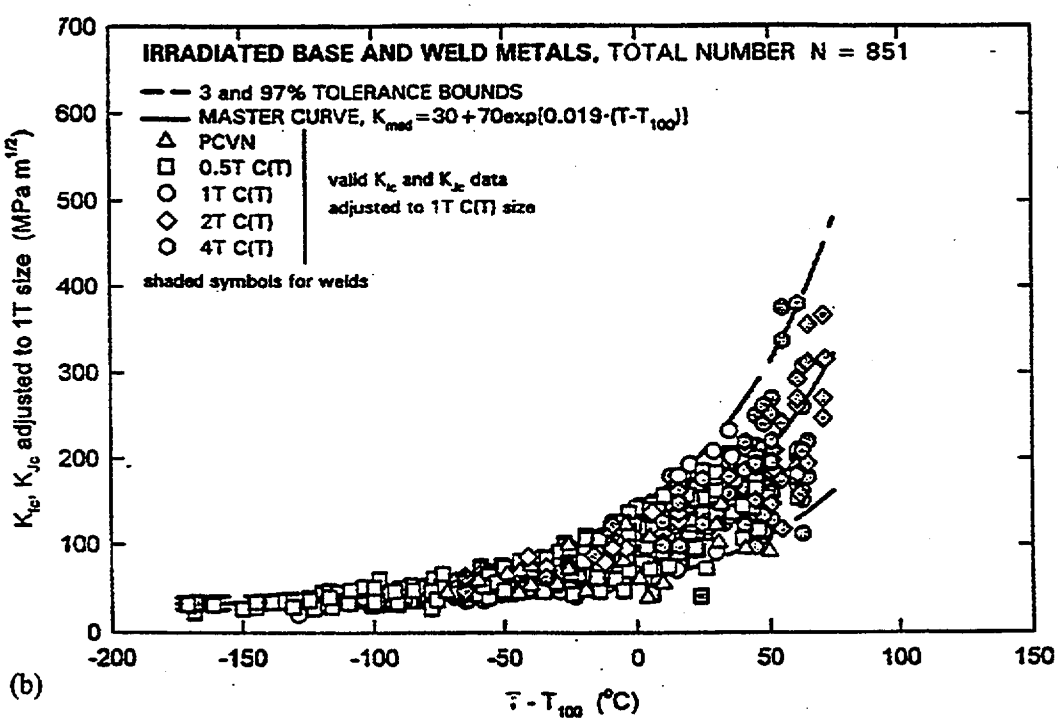
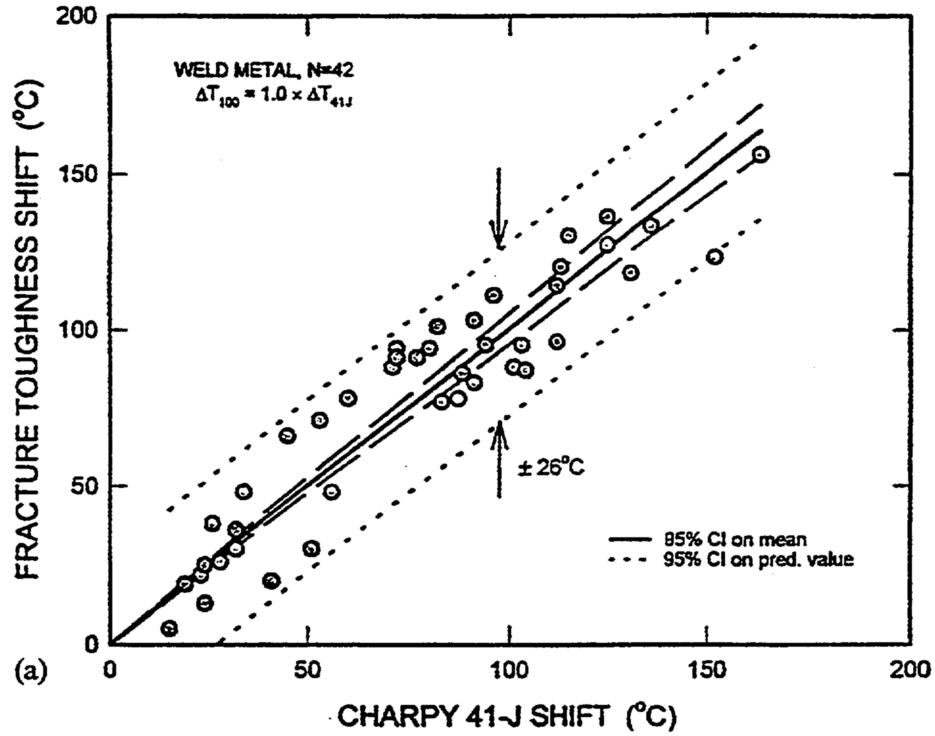


Fig. 14. Plot of fracture toughness data for database of RPV base and weld metals in the irradiated condition showing that the scatter is large but the data are well described by the master curve concept. These results do not suggest a change in shape of the master curve due to irradiation.



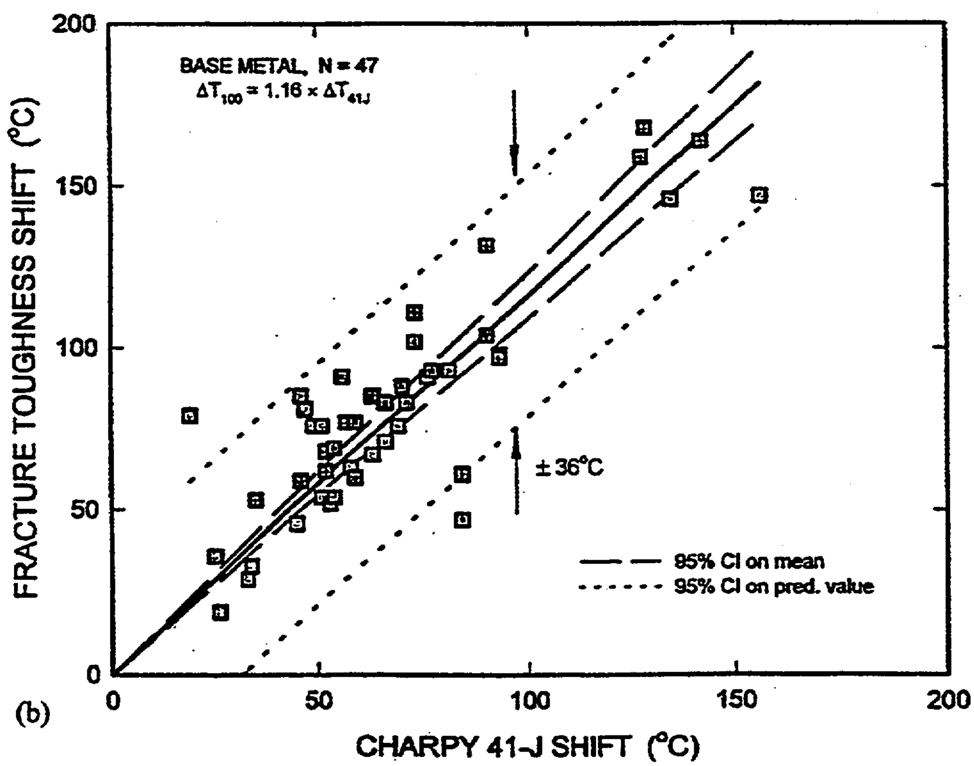
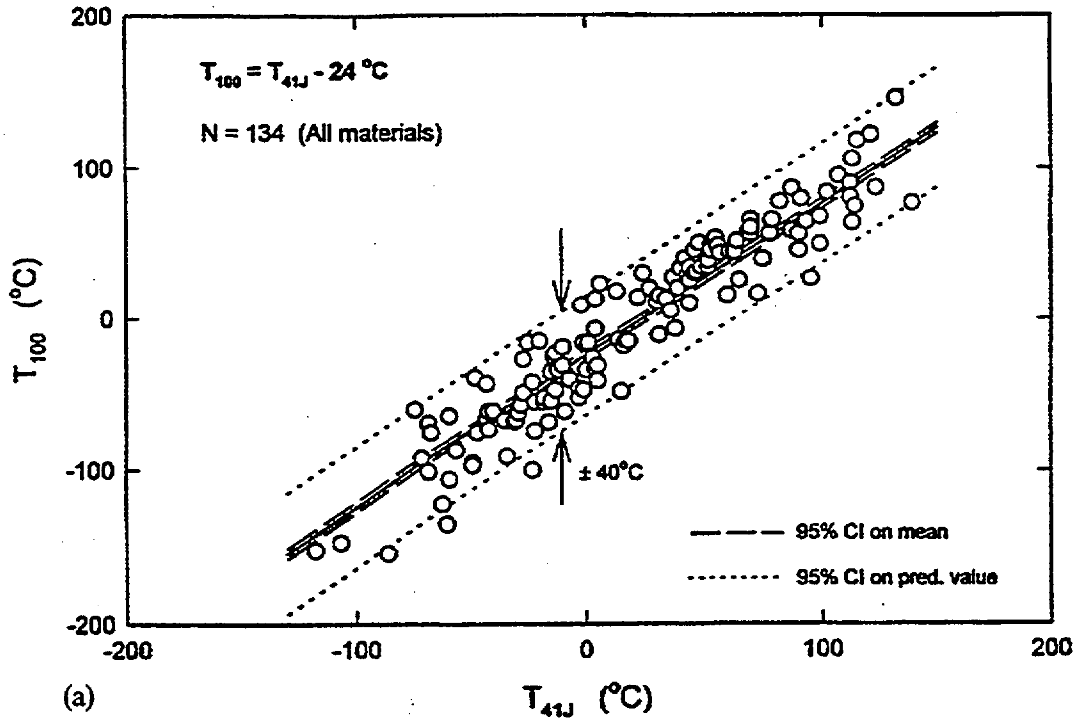
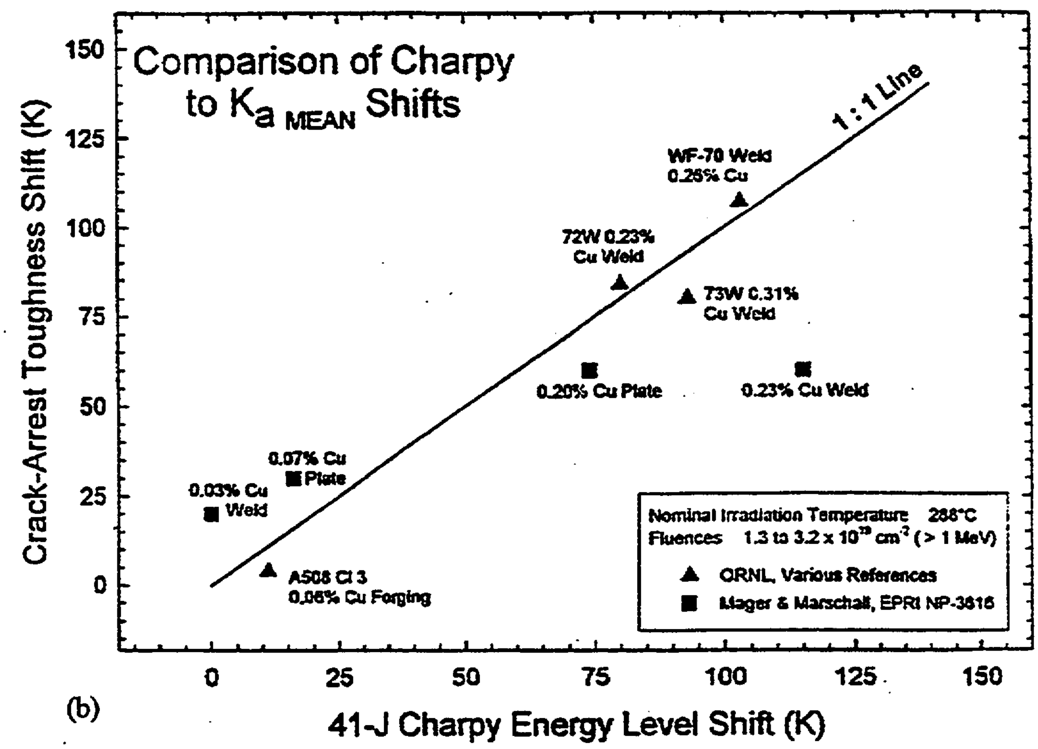


Fig. 15. Comparisons of irradiation-induced fracture toughness and Charpy impact transition temperature shifts for (a) weld metals, and (b) base metals. On average, the shifts are about the same for weld metals but the fracture toughness shift is about 16% greater than the shift for base metals, with very large 95% confidence limits in both cases.



(a)



(b)

Fig. 16. Plots of (a) fracture toughness transition temperature vs Charpy 41-J temperature for all the materials in the database showing a good correlation but with a large confidence interval, and (b) crack-arrest toughness shift vs Charpy 41-J shift showing the shifts are about the same, but for a very sparse database.

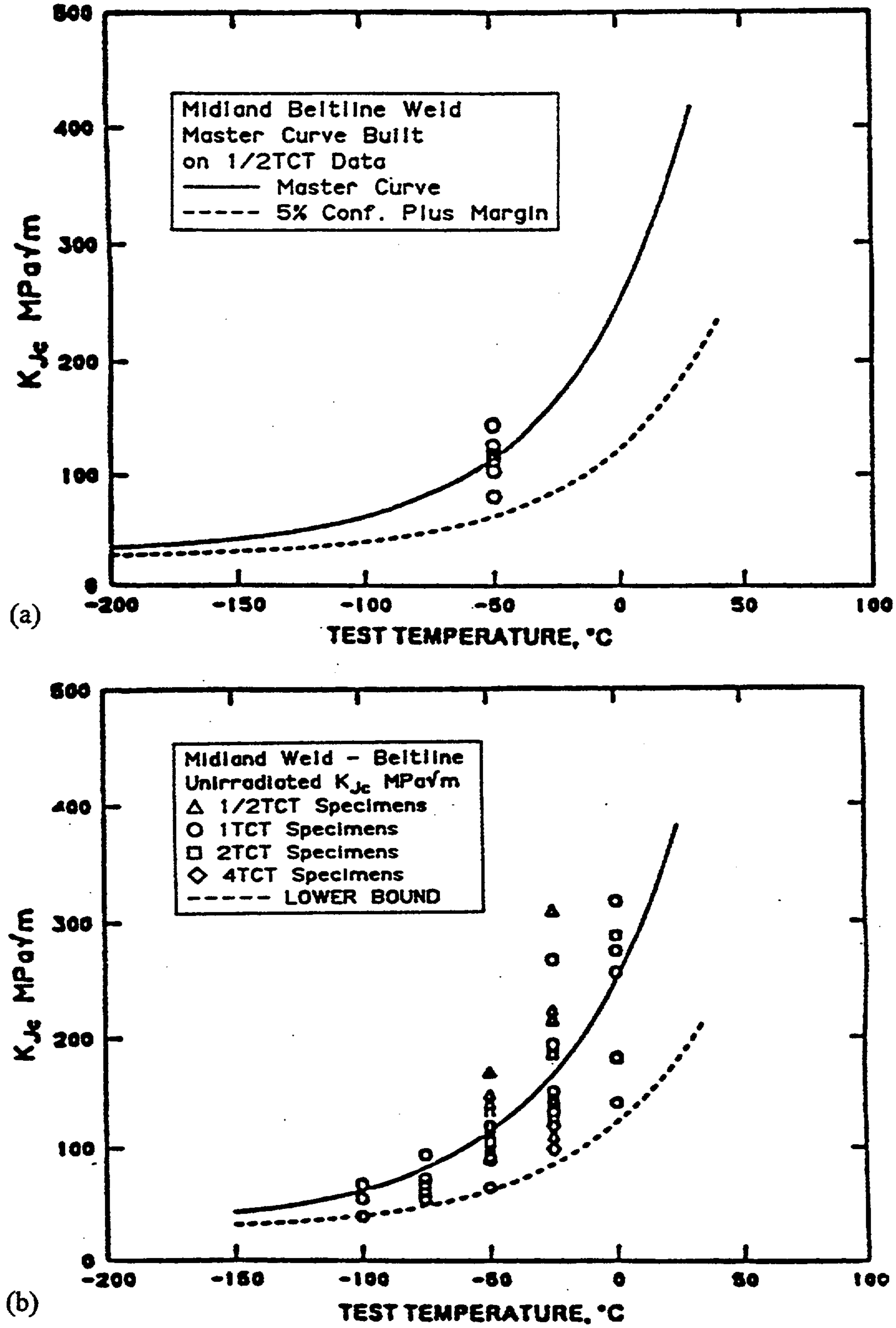
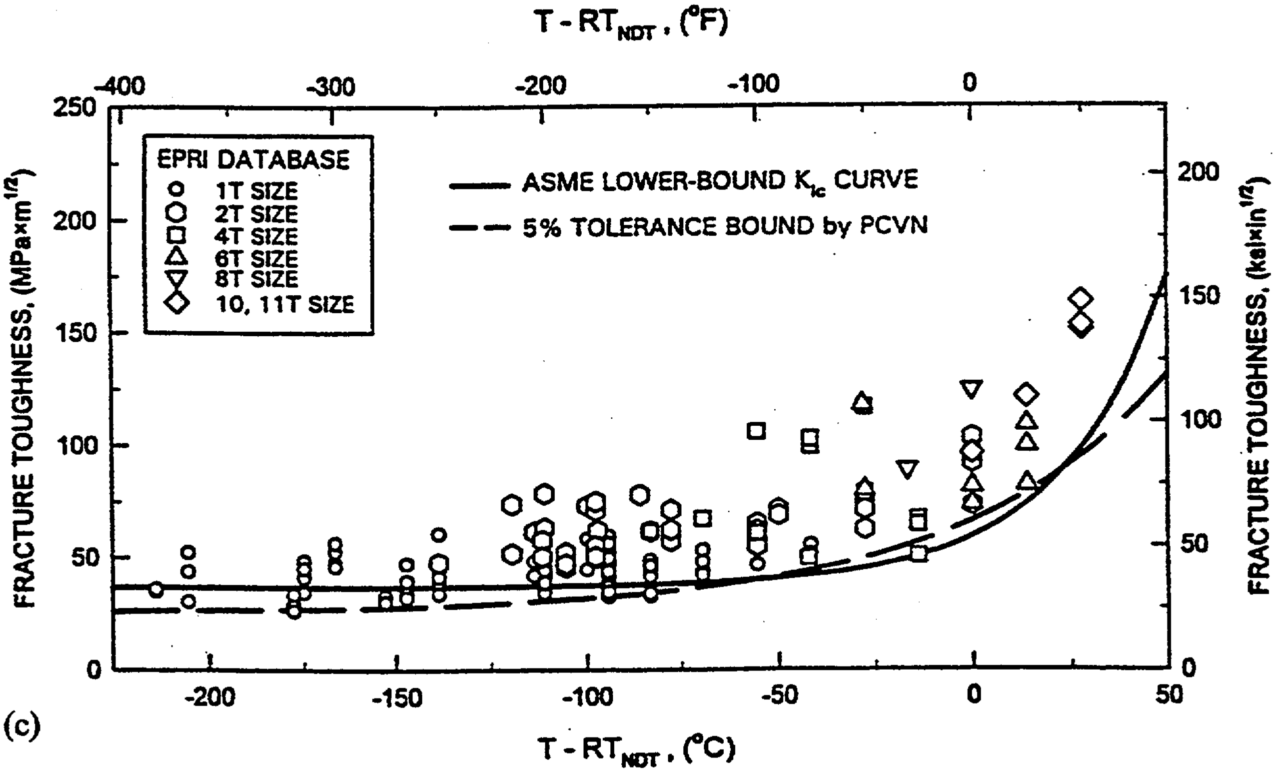


Fig. 17. Plots of fracture toughness vs temperature showing (a) results of precracked Charpy tests for Midland Unit 1 RPV beltline weld with master curve, (b) master curve from the precracked Charpy data relative to all the data for the weld including those from larger specimens, and (c) comparison of the 5% tolerance bound curve based on tests of HSST Plate 02 and the overall fracture toughness database used to construct the K_{Ic} curve in the ASME Code.



(c)

Fig. 17. Plots of fracture toughness vs temperature showing (a) results of precracked Charpy tests for Midland Unit 1 RPV beltline weld with master curve, (b) master curve from the precracked Charpy data relative to all the data for the weld including those from larger specimens, and (c) comparison of the 5% tolerance bound curve based on tests of HSST Plate 02 and the overall fracture toughness database used to construct the K_{Ic} curve in the ASME Code.

Regarding the use of small specimens, particularly precracked Charpy specimens, Fig. 17 provides results of testing for the Midland Unit 1 RPV weld and for HSST Plate 02 which show that the specimen offers potential for direct measurement of fracture toughness from surveillance capsules. Two ongoing projects within the HSSI Program at Oak Ridge National Laboratory (ORNL) include effects of irradiation on the shape of the fracture toughness master curve for highly embrittled steel, and the effects of irradiation, postirradiation thermal treatment, and reirradiation on the propensity for temper embrittlement in heat-affected zones. The procurement, modification, and completed installation of a remotely operated guillotine saw and computer numerically-controlled milling machine in the hot cells at ORNL provides a facility for machining of test specimens from sections of irradiated RPVs. Concurrently, a number of other research programs on radiation effects in RPV steels are also in progress within other organizations in the United States and in other countries.

Research in irradiation effects on fracture toughness is providing information relevant to RPV integrity:

1. Developments in elastic-plastic fracture mechanics have been largely driven by the need for accurate prediction of irradiated RPV behavior.

2. Consensus standards have been developed for determining K_{Ic}, J_{Ic}, J-R curves, K_{Jc}, and K_{Ia} of RPV steels.

3. Major irradiation projects have been completed, providing critical information regarding the fracture behavior of RPV steels under conditions of irradiation, thermal annealing, and reirradiation, to include

effects of copper and nickel content, relationships between Charpy impact toughness and fracture toughness/crack-arrest toughness, stainless steel cladding, and low upper-shelf welds.

4. Techniques have been developed which allow determination of fracture toughness transition temperature using a relatively few number of relatively small specimens.

5. Fracture toughness data have been obtained in sufficient quantity to allow for probabilistic application.

6. The combination of irradiation experiments with modeling and microstructural studies provides an essential element in aging evaluations of RPVs.

7. Relative to PTS analysis, the results are directly applicable as the calculations of failure probability are directly dependent on the initiation and arrest toughnesses of the materials.

However, a number of issues remain regarding effects on fracture toughness of RPVs:

1. Further evaluation of specimen size effects are needed to fully understand the limits of applicability and associated uncertainties.

2. Data are needed on the effects of irradiation/post-irradiation heat treatment on the propensity for temper embrittlement of weld heat-affected zones.

3. Additional experimental evidence is needed to investigate the relevance of the fracture toughness master curve to dynamic fracture, the intergranular fracture mode, and very high embrittlement levels; this includes the potential effects of "late-blooming phases."

4. The existing procedures for determining reference temperatures and associated uncertainties bear reevaluation with regard to the results obtained, including statistical improvements and the logic regarding surrogate materials.

5. Data are needed regarding irradiation effects on high-nickel welds and on high-copper low upper-shelf welds at high fluence.

6. Examination of material from decommissioned RPVs would provide valuable data for through-thickness attenuation of radiation damage and for validation of surveillance data and predictive embrittlement models.

7. Continuing integration of irradiation experiments with modeling and microstructural studies is an essential element in the evolution of improved predictive tools.

References

1. G. D. Whitman, *Historical Summary of the Heavy-Section Steel Technology Program and Some Related Activities in Light-Water Reactor Pressure Vessel Safety Research*, USNRC Report NUREG/CR-4489 (ORNL-6259), Oak Ridge National Laboratory, Oak Ridge, Tenn., 1986.

2. U. Potapovs and J. R. Hawthorne, *The Effects of Residual Elements on 550 °F Irradiation Response of Selected Pressure Vessel Steels and Weldment*, NRL Report 6803, Naval Research Laboratory, Washington, D.C., 1968.

3. R. K. Nanstad et al., "Heavy-Section Steel Irradiation Program on Irradiation Effects in Light-Water Reactor Pressure Vessel Materials," in *Fatigue and Crack Growth: Environmental Effects, Modeling Studies, and Design Considerations*, PVP-Vol. 306, S. Yukawa, D. P. Jones, H. S. Mehta, Eds., American Society of Mechanical Engineers, New York, 1995.

4. M. A. Sokolov, "Statistical Analysis of the ASME K_{Ic} Database," *Journal Pressure Vessel Technology, Transactions of the ASME*, 120, 24-28 (February 1998).

5. R. K. Nanstad, J. A. Keeney, and D. E. McCabe, *Preliminary Review of the Bases for the K_{Ic} Curve in the ASME Code*, ORNL/NRC/LTR-93/15, Oak Ridge National Laboratory, Oak Ridge, Tenn., July 1993.

6. K. Wallin, "The Scatter in K_{Ic} Results," *Engineering Fracture Mechanics*, 19(6), 1085-1093 (1984).

7. M. A. Sokolov and R. K. Nanstad, "Comparison of Irradiation-Induced Shifts of K_{Jc} and Charpy Impact Toughness for Reactor Pressure Vessel Steels," pp. 167-190 in *Effects of Radiation on Materials: 18th International Symposium, ASTM STP 1325*, R. K. Nanstad, M. L. Hamilton, F. A. Garner, and A. S. Kumar, Eds., American Society for Testing and Materials, West Conshohocken, Pa., 1999.

THE EFFECTS OF DEREGULATION OF THE ELECTRIC POWER INDUSTRY ON THE NUCLEAR PLANT OFFSITE POWER SYSTEM: AN EVALUATION

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ABSTRACT

This report presents an evaluation of the impact of electric power industry deregulation on grid reliability and reactor safety. Deregulation has the potential to challenge operating and reliability limits on the transmission system and could affect the reliability of the electric power system and the offsite power to nuclear plants. The report describes the offsite power system, discusses the principal criteria for evaluating the effects of deregulation on the nuclear plant offsite power system, and presents a review of various staff risk-informed and engineering-based initiatives to evaluate deregulation issues related to the nuclear plant offsite power system. This report provides the basis for the information in SECY-99-129, "Effects of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety," May 11, 1999. On the basis of this study, the staff concluded that no further regulatory action was required, but did recommend certain staff follow-up actions. Considered with the U.S. Nuclear Regulatory Commission initiative to evaluate regulatory effectiveness, this report is an example of the Commission evaluating an emerging issue to ensure the regulations would be effective in maintaining an adequate level of safety.

INTRODUCTION

Deregulation of the electric power industry is part of the ongoing national trend to deregulate such major industries as the airlines, telecommunications, and natural gas. Before deregulation of the electric power industry, electricity was generated and transmitted by a single utility within predetermined geographic boundaries. In addition, consumer electricity rates were regulated. Further, a single utility had ownership of the generation and transmission systems that make up the grid in the utility's territory and sole responsibility for the design and coordination of generation and transmission facilities for reliable grid operation. Experience showed that under these conditions, the grid was a reliable source of electric power to the industry's nuclear power plants.

In 1992 the National Energy Policy Act was passed to encourage competition in the electric power industry. The National Energy Policy Act requires, among other things, open access to the electric transmission system without regard to geographic boundaries to promote competition among wholesale purchasers and sellers of electric power and statutory reforms to encourage utility participation in the formation of wholesale generators. The industry started to implement deregulation initiatives in 1996, following the Federal Energy Regulatory Commission (FERC) Order 888, "Promoting Wholesale Competition Through Open Access Non-discriminatory Transmission Services by Public Utilities; Recovery of Stranded Costs by Public Utilities and Transmitting Utilities," April 24, 1996, which called for utilities to assure open and fair transmission access, system reliability, and reduced prices. Although the transmission system will remain regulated, State utility regulatory commissions have deregulated most of the generation system by removing the generating plants from the rate base and opening a power market. As utilities are deregulated they often restructure, creating generation subsidiaries or divesting their generation assets entirely.

Institutional, technical, and operating issues have emerged from the deregulation initiatives. The issues have been identified by FERC, the North American Electric Reliability Council (NERC), and individual licensees. Among institutional issues are those related to FERC requiring national conformance to grid-reliability standards. Among technical issues are those related to changes to the grid design and operating configuration that challenge operating limits and grid reliability. Among operating issues are those related to changes in the ownership, roles, responsibilities, and operational interfaces between the power market, the generating companies, and the transmission system owners. Following Commission briefings by the NRC staff and representatives from the Department of Energy, FERC, NERC, and the industry, the Commission identified four actions for the staff in a staff requirements memorandum (SRM), "Briefing on Electric Grid Reliability, April 23, 1997, and Briefing on Electric Utility Restructuring, April 24, 1997," May 27, 1997 (Ref. 1).

SECY-97-246, "Information on Staff Actions To Address Electric Grid Reliability Issues–WITS [work item tracking system] No. 9700205," October 23, 1997, presented a task action plan in response to the May 27, 1997, SRM. SECY-97-246 reported that three of the actions in the SRM had been completed (make contacts with other agencies; provide information regarding the V.C. Summer pressurized-water reactor (PWR) grid disturbance of July 11, 1989; and make regional contacts with power pool and reliability councils). The NRC also issued Information Notice 98-07, "Offsite Power Reliability Challenges From Industry Deregulation," February 27, 1998 (Ref. 2), to alert licensees to the potentially adverse effect of electric power industry deregulation on the reliability of the offsite power source. The staff completed several activities as part of the task action plan, including a survey of 17 nuclear power plants and electric grid control centers, an assessment of the risk significance of potential grid unreliability due to deregulation, and an evaluation of loss of offsite power (LOOP) events at nuclear plants from 1980 through 1996. Contacts with NERC found that NERC completed reports in 1997 and 1998 assessing future electric power generation and transmission reliability on a regional basis.

This report presents background information for understanding the potential impact of deregulation of the electric power industry on the nuclear plant offsite power system and presents information compiled as part of the staff's task action plan to develop recommendations regarding the fourth item in the SRM:

> *The Commission asked the staff to give greater urgency to ensuring that related health and safety issues within the NRC's jurisdiction are addressed, particularly in reviewing the terms of the licensing basis and validating assumptions about grid reliability.*

2 BACKGROUND

2.1 Description of the Offsite Power System

The offsite power system is the preferred source of ac electric power for the industry's nuclear power plants. Nuclear power plants use this power to start and run redundant ac safety loads (emergency systems and engineered safety features) required to shut down the plant under all conditions.

Accident sequences at nuclear power plants have been initiated by grid disturbances that cause a LOOP to the ac electric safety loads needed to shut down the reactor. A LOOP is an event that occurs when all sources of offsite power are unavailable, causing the ac safety buses to de-energize, and onsite ac emergency power supplies to start and load. NUREG/CR 5496, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996," June 1998 (Ref. 3), shows that there were 173 LOOPs between 1980 and 1996. Review of the accident sequence precursor (ASP) database found that 71 of the LOOPs met or exceeded the ASP threshold of 1E-6 conditional core damage (CCDP) probability. Plant-specific probabilistic risk assessment (PRA) studies have shown that station blackout (SBO) can be a significant contributor to core damage frequency (CDF). An SBO results from a LOOP and from the loss of all onsite ac emergency power. The range for the frequency of core damage as a result of an SBO accident is estimated in NUREG-1032, "Evaluation of Station Blackout Accidents at Nuclear power Plants," June 1988 (Ref. 4), as 1E-4 to 1E-6 per reactor-year. NUREG-1109, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout," June, 1988 (Ref. 5), states as a goal that the expected CDF from an SBO could be maintained around 1E-5 or less per reactor-year for almost all plants.

By design, the offsite power system should be robust enough not to cause a LOOP or voltage or frequency instability following a trip of the nuclear unit or the largest single load or generator on the power system. In addition, the capacity and capability of the offsite power system are frequently demonstrated during a transfer of the station loads from the unit auxiliary transformer to the offsite power supply following a unit trip. This is a severe test of the offsite power system since the unit trip significantly reduces the power being delivered to it and, within a few seconds, requires additional offsite power to run the station loads. The adequacy of the offsite power system to start and run the safety loads is demonstrated since the station load transferred generally exceeds the total accident safety load by a factor of 5 to 10.

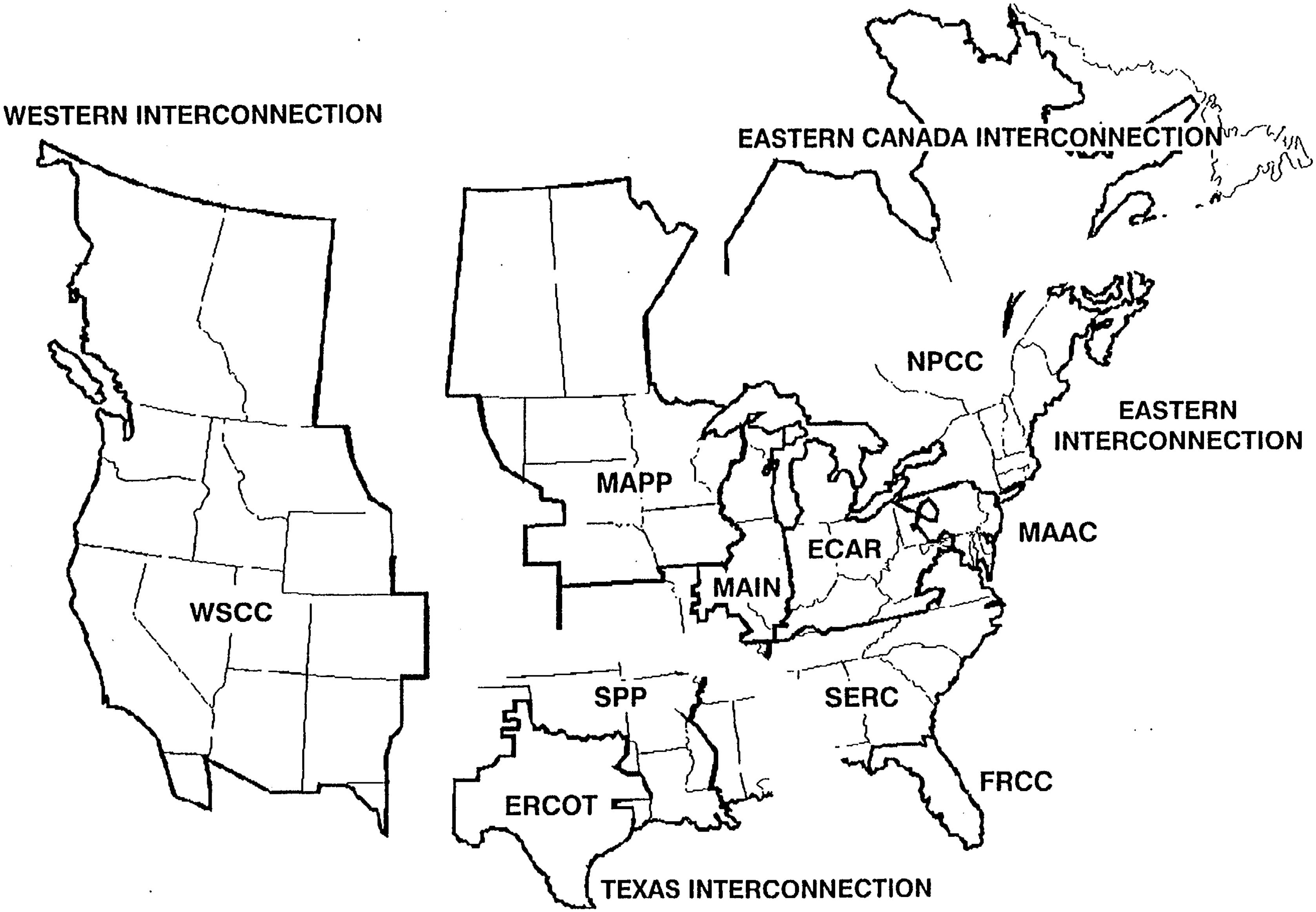

FIGURE 1 – Major Interconnections and NERC Regional Councils

"Grid reliability" is presently controlled nationally through voluntary participation in the NERC. NERC is a consensus organization that has 10 regional councils and a large internal board of directors. Figure 1 shows the geographic areas that correspond to the 10 NERC regional councils. NERC is planning to reorganize as the North American Electric Reliability Organization. The North American Electric Reliability Organization is planned to be a self-regulating organization, with a smaller external board of directors, whose powers are defined by FERC. The North American Electric Reliability Organization is expected to require national conformance to a set of reliability standards.

2.2 Principal Criteria for Evaluating the Effects of Deregulation on the Nuclear Plant Offsite Power System

From an engineering, licensing, and risk perspective, evaluation of the effects of deregulation of the nuclear plant offsite power system involves the criteria in Table 1, "10 CFR 50 Offsite Power System Principal Criteria and Common Measures." Table 1 shows the principal offsite power system design criteria and their common measures. The principal design criteria, including both deterministic and risk considerations, that provide the licensing basis for the offsite electric power system appear in 10 CFR 50, Appendix A, General Design Criterion (GDC) 17, "Electric Power Systems." GDC 17 establishes the following minimum requirements for the principal design criteria for the offsite electric power system: offsite power system capacity, capability, availability, and provisions to minimize the probability of a LOOP. In 10 CFR 50.63, "Loss of all alternating current power," the staff establishes that the SBO duration shall be based on the reliability of onsite emergency ac power sources, the expected frequency of a LOOP, and the probable time needed to restore offsite power. Licensees generally address conformance of the offsite power system to these requirements in their FSARs. GDC 17 and 10 CFR 50.63 establish other requirements, such as the number of offsite connections to the plant; these are not listed in Table 1 since they are fixed by the design.

Evaluation of the principal criteria in Table 1 is consistent with Oak Ridge National Laboratory (ORNL)/NRC/LTR/98-12, "Evaluation of the Reliability of the Offsite Power Supply as a Contributor to Risk of Nuclear Power Plants," (Ref. 6) which recommends that offsite power [design] basis requirements be adequately addressed.

Table 1 10 CFR 50 Offsite Power System Principal Criteria and Common Measures

Principal Criteria	Measure
Capacity	MVAR and MW generation and load mismatches that degrade offsite power system frequency and voltage
Capability	Voltage Frequency
Provisions to minimize LOOP probability	A unit trip should not degrade voltage to initiate a LOOP
Availability	Duration of a LOOP Expected frequency of a LOOP
SBO duration basis	Reliability of the onsite emergency ac power sources Expected frequency of a LOOP Probable time needed to restore offsite power

2.3 The Potential Impact of Deregulation on the Nuclear Plant Offsite Power System

The following documents indicate that deregulation has the potential to adversely impact grid reliability: NERC grid reliability forecasts in NERC "Reliability Assessment, 1997–2006" (Ref. 7) and "Reliability Assessment, 1998–2007," (Ref. 8); an NRC site survey in ORNL/NRC/LTR/98-12 (Ref. 6); the actions of the California Independent System Operator as obtained from a site visit; and a general stability assessment by the Electric Power Research Institute (EPRI) (Ref. 9). This information was used to identify the following potential impacts of deregulation of the electric power industry on the offsite power system that could affect the principal criteria in Table 1:

(1) The grid design and operating configurations were established before the electric power industry was deregulated to ensure the correct voltages and frequencies on the system. Deregulation may result in unanalyzed grid operating configurations from (a) daily changes in the operable generators from implementing the power market and (b) power load flows changes from the consumer's selection of a supplier, which identifies where the power flows. Once the circuit configuration is defined, Kirchhoff's laws of electricity, not the power market or consumers, determine how the current divides among the different grid paths under each operating configuration. Failure to analyze the grid under changing conditions and reconfigure the grid to avoid adverse configurations could result in the following:

- Transmission line congestion, that is, individual transmission current flows that exceed previously established limits and cause abnormal voltages at the nuclear plant(s) while the plants are operating.

- Unexpected responses of the grid following faults on the generation or transmission system that cause abnormal voltages and frequencies at the nuclear plant(s).

(2) Defaults on generation bids may erode reserve capacity margins that are needed to maintain system frequency and voltage stability following a disturbance.

(3) Assumptions about the availability of the offsite power supplies could change. Licensees are selling their generating facilities that supply offsite power to the nuclear plants. In some cases, licensees are selling the black-start power supplies that are used to restore power to the grid following a grid blackout.

(4) The duration of a LOOP or an SBO may increase. Changes in ownership and control of generation and transmission facilities may increase recovery time because of less coordination between generation and transmission facilities following a grid disturbance.

(5) The NERC reliability forecasts and the Office of Nuclear Reactor Regulation (NRR)/ORNL site survey show that the effects of deregulation on the nuclear power plants are regional. A major grid disturbance could affect several nuclear plants simultaneously.

(6) Pressures to keep electricity rates competitive may result in a reduction of grid maintenance or a reluctance to invest in transmission system upgrades that are needed to preserve the present level of grid reliability to the nuclear plants.

(7) As nuclear units are sold to nonutility entities, the new owners may choose to operate differently to compete in the power market. For example, nuclear generators may need to load-follow, that is, run fully loaded during the week days and partially loaded at other times. This could potentially impact the licensee, reactor systems, and fuel performance.

2.4 Scope

The following items are discussed in Section 3 and provide the basis for the staff's recommendations regarding deregulation:

- The operating experience was assessed to identify and evaluate potential weaknesses regarding (1) previous protective schemes that could complicate power system availability and reliability and (2) the provisions to minimize the probability of a LOOP.

- NERC reliability forecasts were used to obtain insights on future generation system adequacy and transmission system security.
- Sensitivity studies were reviewed for potential changes to event frequency and duration related to SBO.
- The operating experience was used to verify that the reliability of the onsite emergency power system is as assumed in the analysis of the risk margins for LOOP and SBO events.

Potential concerns from weather-initiated and nongrid-initiated LOOP events, and the reliability of systems, such as high-pressure coolant injection, needed to cope with an SBO, were not included in the scope of the study.

3 EVALUATION OF THE PRINCIPAL CRITERIA

This section of the report uses the criteria and measures in Table 1 to evaluate the potential effects of deregulation of the electric power industry on the nuclear plant offsite power system. The principal criteria were grouped into the following three sections: (1) the adequacy of the offsite power system voltage and frequency at the nuclear plants was used to evaluate capacity and capability, (2) minimizing the probability of a LOOP following a unit trip, and (3) risk and reliability measures were used to evaluate availability and SBO duration basis. Each section ends with an evaluation and recommendation that is used to formulate a conclusion and recommendations at the end of the report.

3.1 Adequacy of the Offsite Power System Voltage and Frequency at the Nuclear Plants

NRC requirements address the adequacy of voltage measures and require degraded voltage protective devices to assure that the requirements GDC 17 in Appendix A to 10 CFR 50, are satisfied.

In a letter sent to all power reactor licensees, "Adequacy of Station Electric Distribution System Voltages," August 8, 1979 (Ref. 10), the NRC required analysis to confirm the adequacy of the voltage levels in the station's electric distribution system. In summary, the letter calls for plant-specific analysis, which shows that the offsite power system and the onsite distribution system are of sufficient capacity and capability to start and operate redundant ac safety loads within their required voltage ratings in the event of an anticipated transient (such as a unit trip) or an accident (such as a loss-of-coolant accident), whichever presents the greater load. Evaluation of the adequacy of the grid as a source of offsite power for a nuclear plant requires analysis of a circuit whose components are the nuclear plant electrical distribution system, the offsite and onsite generators, and the components of the transmission system. The analysis generally results in a worst-case minimum and maximum plant voltage, each based on the worst-case offsite power system capacity (MW and MVAR) and voltage conditions. The analysis also results in alarm and protective setpoints for required degraded voltage protective devices. The NRC reviewed the licensee responses to the August 8, 1979, letter. In addition, the NRC reviewed these analyses as part of an Electrical Distribution System Functional Inspection at some licensee facilities from 1989 to 1992.

Regardless of the outcome of deregulation of the electric power industry, nuclear plant protective features ensure protection of ac safety equipment from abnormal offsite power system voltage and frequency resulting from deregulation and all other conditions. Operation of degraded voltage relays initiates the following: (1) isolation of the offsite power system from the onsite ac electric safety-related auxiliaries, (2) the start of the onsite emergency ac power supplies, and (3) loading of the ac safety-related loads to the onsite supply. These relays operate as a result of a blackout or any other condition that degrades voltage, including conditions that may be caused by deregulation initiatives. A low-voltage alarm alerts operators to declining voltage conditions. However, numerous licensee event reports from 1993 to 1998, indicate weaknesses that affected the adequacy of the degraded voltage design, particularly the degraded voltage protective setpoints, and in the administrative controls to cope with the LOOP. Such weaknesses may also indicate weaknesses in the technical guidance or that the licensees are periodically reviewing potential degraded voltage issues. Operating experience indicates that the existing technical guidance on offsite power system voltage issues, including the degraded voltage relay setpoints, needs to be addressed and was included in recommendation 2a (Ref. 11).

A degraded grid that affects the nuclear plant ac safety loads may also result from abnormal frequencies. At most nuclear plants, there is no frequency protection to isolate the nuclear plant's safety buses from abnormal offsite power supply frequency. However, plant and offsite power system protective relay operations are coordinated to ensure that the system frequency does not drop below acceptable levels, and to prevent widespread blackouts. These relays automatically trip unstable system generators and system loads, in a deliberate manner, until the system stabilizes. These relays operate as a result of insufficient capacity, blackout, or any other condition that leads to degraded frequency, including conditions caused by deregulation initiatives. However, the Western Systems Coordinating Council (WSCC) identified design deficiencies and recommended attention in the areas of the automatic controls and the protective relaying that ensure the adequacy of the system frequency (WSCC, 1994) (Ref. 12) and (WSCC, 1996) (Ref. 13). The WSCC investigation of major grid disturbances that occurred in 1994 and 1996 found that devices did not operate properly to sectionalize the grid to maintain frequency stability, resulting in the cascaded trips of several nuclear and fossil units that exacerbated the disturbance. In addition, a site survey of 17 plants recommended the following: "NRC staff should reevaluate the underfrequency protection trip settings and other grid considerations in view of concerns regarding cascading trips" (Ref. 8). This recommendation was included in SECY 99-129 (Ref. 8) under recommendation 2a. This includes the protection of the ac safety loads as, and whether, the existing reactor coolant pump underfrequency protection, which trips the unit when reactor limits are exceeded, is coordinated with the other frequency protection and leading to cascading trips during grid events. The recommendation was based on information from a utility suggesting that the independent system operator (ISO) protocol should require rigorous analysis of selected underfrequency scenarios.

A search of the operating experience also found that grid transients may initiate reactor protective trips before the event reaches the threshold of a degraded grid or LOOP. The search did not find any weaknesses in reactor protective features as a result of grid transients.

Evaluation and Recommendation

The nuclear power plants and the grid, in combination, must have adequate voltage and frequency protective relaying and alarms to ensure that changes in the design and operation of the offsite power system in a deregulated environment do not result in abnormal levels of voltage or frequency at redundant ac safety-related loads under any condition. In this regard, there are regulatory controls in place to ensure the adequacy of protective relays for emergency buses. However, the operating experience indicates weaknesses that affected the degraded voltage protective setpoints and scope of the frequency protection that need to be addressed.

To ensure that plant ac safety loads are protected from abnormal offsite system voltages and frequencies the staff will evaluate the adequacy of (1) the existing technical guidance on offsite power and voltage issues, (2) the degraded voltage relay setpoints, and (3) the scope of the offsite power system frequency protection to include whether the existing grid and reactor coolant pump underfrequency protection coordination could lead to unnecessary trips.

3.2 Minimizing the Probability of a Loss of Offsite Power Following a Unit Trip

In 10 CFR 50, Appendix A, GDC 17 requires in part that provisions shall be included to minimize the probability of losing electric power from any of the remaining offsite power supplies as a result of, or coincident with, the loss of power generated by the nuclear unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies. Even without the requirements in GDC 17, most utilities design and operate the offsite power system so that a unit trip (of any fossil, hydro, or nuclear unit) does not result in significantly degraded voltages or frequencies. To comply, the grid must have sufficient grid reserve capacity margin (MW and MVAR) to accommodate a nuclear plant trip that reduces the grid MW and MVAR capacity, and within a few seconds requires additional MW and MVAR to start and run the nuclear plant's ac auxiliary and safety loads. Review of several FSARs indicates that licensees have generally stated therein that a unit trip will not result in a LOOP. However, review of the operating experience from 1988 through 1998 found five events where a LOOP followed a unit trip. Review found five cases in which a unit trip caused a voltage drop or loss of voltage to the ac safety loads at a nuclear plant. The staff noted that the V.C. Summer unit trip in

1989 demonstrates how a unit trip resulted in cascading trips of other units, grid instability, and a LOOP.

Evaluation and Recommendation

Collectively, five unit trips over a 10-year period is not a significant number. However, individually, the trips may indicate specific offsite power system weaknesses that resulted in a LOOP at the nuclear plants.

A LOOP following a unit trip should be investigated by the staff as a degraded voltage issue.

3.3 Reliability and Risk

The assessment for future grid reliability and risks involves unquantifiable amounts of uncertainty as follows:

- The NRC and licensee reliability and risk assessments are based on historical operating data in a regulated industry and generally show that the nuclear plant offsite power supply is reliable. Grid operation in a deregulated industry may result in different operating data, particularly in view of changes in operation of the grid due to new roles and responsibilities, new entities that include the power exchange, and different operating reliability and engineering limits criteria that have the potential to invalidate the historical data.

- NERC grid reliability forecasts and an ORNL/NRC evaluation of the offsite power system as a contributor to risk at nuclear power plants used forecasted data developed by industry experts. The NERC reliability assessments have stated that assessing reliability beyond the near term is extremely difficult because of the level of uncertainty in the data since neither the generation resources nor loads are known in an open market.

It is important to understand (1) how the data for LOOP events are collected, and (2) that changes are being made to ensure that all LOOP events are reported to the NRC. LOOP events are generally reported to the NRC in accordance with 10 CFR 50.72 and 50.73 as a result of engineered safety features actuations that result in the start and loading of onsite emergency ac power supplies. However, there are seven plants at which the emergency power supplies are not considered engineered safety features. The staff issued the proposed rule revisions to capture these seven plants in accordance with 10 CFR 50.72 and 50.73 in June 1999, and plans to issue the final rule in February 2000.

It is also important to understand how the NRC data need to be used to capture all grid-initiated events. NRC studies generally classified LOOP events into severe-weather-related events, grid-related events, and plant-centered events. Severe-weather-related events are those in which the weather affected a large area and are capable of disrupting plant operation. Grid-related events are those in which widespread power system problems in the offsite power grid caused and impacted the duration of the LOOP. The data in NUREG/CR-5496 indicate six grid-related events. Plant-centered events are those in which the plant design and operational factors played a major role in the cause and duration of the LOOP. There are plant-centered grid-initiated events within the plant-centered grouping that had a local effect on the grid, such as transmission system load dispatcher errors, transmission line faults near the plant, and switchyard events. The data in NUREG/CR-5496 show there were six plant-centered, grid-initiated events to include one load dispatcher error and five transmission line faults near the plant. In total there were 12 grid-initiated events and these were counted in the RES study that is discussed in Section 3.3.5. The 32 LOOP events initiated in the plant switchyard, which is located on the plant site but is part of the transmission system, were not counted for the purposes of risk studies. Understanding how the events are classified may be important when discussing grid failures with NERC, FERC, and the Department of Energy.

3.3.1 North American Electric Reliability Council Assessments of Future Grid Reliability

NERC presently considers the potential impact of deregulation and forecasts the generation and transmission system reliability on a regional basis. NERC has completed three reliability assessments: (1) "Reliability Assessment, 1997–2006," October 1997, (2) "Reliability Assessment, 1998–2007," September 1998, and (3) "1998 Summer Assessment," May 1998. Major observations from review of the reports follow:

- NERC reliability forecasts state that the system will be adequate for the next 3 to 5 years, but it faces significant challenges in transition to a fully competitive and open market.

- NERC reliability forecasts are highly area sensitive. NERC regional self-assessments present specific regional reliability concerns and plans or completed actions that address the concerns.

- NERC has identified opportunities for improvement, areas for increased attention, and the need to monitor performance.

NERC predictions about deregulation reducing capacity margins materialized in the summer of 1998 for some licensees.

- Detroit Edison, Philadelphia Electric Company, and Enron have filed complaints that as a result of new FERC procedures, some companies cut off power deliveries from their competitors by citing the risk of overloading their own transmission lines. The new procedures allow a utility to disconnect other utilities from its system if it finds that the reliability of the system is being endangered by the connection (Wall Street Journal, July 24, 1998).

- One power trader defaulted on its contract to supply electricity to several utilities, (Southern Company and First Energy Corporation, holding company for Duquesne, Centerior, and Ohio Edison power companies) following a sharp increase in power market prices (Wall Street Journal, July 9, 1998).

3.3.2 Evaluation of Loss of Offsite Power Events

The staff developed NUREG-1032 as part of its resolution of Unresolved Safety Issue A-44, "Station Blackout." In NUREG-1032, the staff discusses actual LOOP events that took place from 1968 through 1985. These were divided into three categories: plant-centered, weather-related, and grid-related. The staff found that frequency and duration of a LOOP are (and remain) important aspects to SBO accident sequences that can dominate the total risk at some nuclear power plants. These data are updated in NUREG/CR-5496.

Table 2, "NUREG-1032 and NUREG/CR-5496 – LOOP Frequency and Duration Comparison," shows that for grid-related LOOP events, the initiating frequency has a decreasing trend (an approximate order of magnitude reduction), and the duration has an increasing trend (an approximate increase by a factor of 4). NUREG/CR-5496 notes that the recovery times tend to be longer, but the data set is small.

Table 2 NUREG-1032 and NUREG/CR-5496 LOOP Frequency and Duration Comparison

Type of LOOP event	Frequency of occurrence per reactor-year		Median duration in minutes	
	1032	CR-5496	1032	CR-5496
Grid-related	0.018	0.0019	36	140
Plant-centered	0.087	0.04	18	20
Weather-related	0.009	0.0066	270	144
Total	0.114	0.0485	—	—

The data in NUREG-1032 and NUREG/CR-5496 also show that plant-centered events were the major cause of LOOP events, and that weather-related and grid-related events caused LOOPs to a much lesser degree. The data show that grid-related events are only a small percentage of the total LOOP frequency per site-year that is used as an input to PRAs that evaluate accident consequences from total LOOP. In addition, considering all the data in NUREG-1032 and NUREG/CR-5496, the median national average duration of LOOPs is approximately 30 minutes. The data in NUREG/CR-5496 reveal that if 10 additional grid-related events and plant-centered grid-initiated events were to occur, it would change the most recent total LOOP initiating frequency from 0.0485 to only 0.0586 per reactor-year. An order-of-magnitude increase in the grid-related and plant-centered grid-initiated LOOP initiating frequency would not result in a significant change in the total LOOP initiating frequency used in the PRA. Consequently the potential increases in risk due to deregulation is likely to be low.

The letter that issued NUREG/CR-5496 (Ref. 14), recommended that no further regulatory action is needed with respect to milestone 4 of the task action plan on grid reliability. The AEOD letter also stated that increases in grid-related LOOP frequency can be identified through routine monitoring and analysis of operating experience before they become a significant contributor to risk from LOOP events. In addition, the grid-related LOOP duration has an increasing trend that can also be identified through routine monitoring and analysis of operating experience before becoming a significant contributor to risk from LOOP events. Further, LOOP events above the national median duration of 30 minutes may need review through the inspection process.

3.3.3 Accident Sequence Precursor Evaluation of Grid-Initiated Loss of Offsite Power Events

Offsite power system disturbances and LOOPs that initiate accident sequences are evaluated on an ongoing basis as part of the NRC ASP program. An ASP event has a CCDP of 1.0E-6 or more. The staff reviewed ASP results for grid-related and plant-centered grid-initiated events from 1980 to 1996 that were also grid-initiated LOOP events. The review found the six grid-related events that occurred from 1980 to 1996. ASP results for plant-centered grid-initiated events from 1980 to 1996, found nine plant-centered, grid initiated events that were ASP events. The review included events in the switchyard that are part of the transmission system and that often had plant involvement through the operator interface, and three events were counted twice since they affected two units. The reviews also show the following:

- All grid-related events were ASP events.

- There has been, on average, one grid-initiated ASP LOOP event per year from 1980 through 1996.

3.3.4 Site Visits To Evaluate the Reliability of the Offsite Power Supply as a Contributor to Risk at Nuclear Power Plants

Members of the staff, with contractor support from ORNL, visited 17 nuclear power plants and their associated system control centers to obtain information regarding system operation during the transition to a deregulated environment. These visits included at least one plant in each of the 10 regional councils that are members of NERC. A standard set of questions was asked at each site visit. ORNL/NRC/LTR/98-12, (Ref. 6) analyzes information obtained during the visits and documents a wide range of concerns from weaknesses in addressing the impact of deregulation.

As part of the ORNL/NRC/LTR/98-12 report, a set of criteria was developed to gain a subjective method for quantifying the future impact of electric industry restructuring on LOOP frequency and time to restore offsite power. Expert opinion was used to apply the criteria to individual nuclear plants and to provide (1) a set of multipliers to be applied to the LOOP frequency developed from NUREG-1032 and NUREG/CR-5496, and (2) revised times to recover offsite power. This method is discussed in Section 5 of the ORNL report, and was applied to a group of 17 plants. Plants at which ORNL identified potential concerns with the transition to a deregulated environment were assigned a multiplier that increased their LOOP initiating frequency or regional blackout recovery time. Conversely, plants that had analyzed or were analyzing the transmission system to ensure adequate voltage were assigned multipliers that decreased their LOOP initiating frequency. Also, plants that had well-defined grid blackout procedures were assigned multipliers that decreased their regional recovery time.

The multipliers for the LOOP frequency ranged from 0.5 to 3.4, with an average of 1.0. The LOOP frequency for 4 of the 17 plants exceeded the average. The predicted time to restore offsite power ranged from 0.2 to 5.1 hours, with an average of 1.9 hours. Seven of the plants had recovery times that exceeded the average. Of the 17 plants, 3 were assigned multipliers that increased above the average both the LOOP initiating frequency and time to recover.

The following were noted from the review of the ORNL and NERC reports:

- Onsite follow-up found there is significant diversity among NERC regions and between utilities within regions in addressing the potential effects of deregulation that may impact the risk that are not evident from risk analyses. For example, predicted increases in the frequency and duration at some plants indicate that not all licensees will address deregulation without potentially eroding risk margins, and this is in conflict with a previous statement that the potential increases in risk due to deregulation is likely to be low.

- ORNL/NRC/LTR/98-12 indicates NERC regional areas of concern. The ORNL report has identified grid-reliability concerns in some of the same areas as NERC reliability assessments. However, the NERC reliability assessments include regional self-assessments that generally provide completed or planned actions to address the concerns.

- The ORNL protocol that was used to conduct interviews at the sites and control centers, if updated to address the ORNL concerns, could be used as a guideline for NRC follow-up of future grid events as appropriate.

The following recommendations are made in ORNL/NRC/LTR/98-12. The staff's disposition of these recommendations is also discussed in Section 3.3.7.

- NRC staff should consider the need to have nuclear power plants confirm that their offsite power [design and licensing]-basis requirements are being adequately addressed.

- The impact of restructuring across the Nation in the next 5 years will most probably be significant; but currently, local area impacts are difficult to anticipate until ISO alignments and industry structure are fully established. The NRC should be vigilant to ensure that the system planning and operating rules and all proposed rule changes at the national, regional, and local levels do not significantly reduce the reliability of offsite power to nuclear power plants.

- NRC staff should reevaluate the underfrequency protection trip settings and other grid considerations in view of the concern regarding cascading trips.

The staff visited the California ISO in May 1998 and March 1999 as part of the staff action plan. The California ISO was of particular interest since California has fully deregulated, and consequently, is one of the areas that has addressed grid-reliability issues resulting from deregulation. The California ISO is a nonprofit agency that assumed operation of the California (and nearby) transmission systems from investor-owned utilities on March 31, 1998, as part of deregulation of the electric power industry in California. Like the power pools in the U.S., the California ISO manages and controls regional operational and engineering activities related to maintaining grid reliability. Unlike other regional grid-reliability centers, the California ISO is mandated by a state law (AB 1890) that mentions reliability 26 times and gives the California ISO powers to address grid reliability.

The California ISO addressed the adequacy of the grid and nuclear plant ac power systems in terms of the factors that drive reliability, minimize power interruptions, and facilitate recovery. At a meeting with NRC in March 1999, the California ISO stated that the greater command, control, and communication within the WSCC was a significant contributor to enhancing grid reliability. The following actions also significantly enhance command, control, and communication, and thus grid reliability:

- The NERC/WSCC grid reliability standards were revised for reliable operation of the grid as a result of events that had adverse effects on the adequacy and security of the western interconnection.

- Transmission control agreements were established between the generator and transmission system owners as binding contracts that specify technical and administrative terms and conditions to help ensure grid reliability.

- The California ISO performs the long-term, annual, daily, and hourly electrical security analysis to ensure that power system is operated in an analyzed configuration.

- The California ISO provides the generators and transmission system owners with daily and hourly watt, volt amperes reactive, and voltage requirements. The California ISO coordinates generator and transmission owner outages and redirects the scheduling to meet the requirements.

- Approximately $400 million was spent to conceptualize, plan, design, build, and implement the technical and operational processes and the monitoring, dispatch, and communication systems to ensure reliable operation of the grid.

- The California ISO has the authority to implement emergency procedures (i.e., for emergency market intervention) to redirect units on, loads off, and purchases/ sales/ resales. The California ISO reviewed the adequacy of the restoration and recovery procedures from a grid disturbance, particularly at licensees that have divested their offsite power supplies, or when the licensee no longer directly operates the transmission and generation systems.

3.3.5 Risk Significance of Potential Grid Unreliability Due to Deregulation

"Risk Significance of Potential Grid Unreliability Due to Deregulation," (Ref. 15) was completed by the Probabilistic Risk Analysis Branch, RES, to analyze the risk significance of grid unreliability as part of the task action plan. This sensitivity study was based on the postulated frequency of LOOPs and recovery times developed in ORNL/NRC/LTR/98-12 and on the data and models in NUREG/CR-5496 to include both grid-related events and plant-centered, grid-initiated events. The analysis estimated the increase in CDF caused by deregulation of the average plant (i.e., if all plants had the same risk) and for outlier plants (i.e., plants that might be most affected by deregulation). For the outlier plants, the maximum increase in CDF was based on the maximum increase in LOOP frequency and the worst case identified in ORNL/NRC/LTR/98-12. The RES study concluded that the risk significance of potential grid unreliability due to deregulation is likely to be minimal for the average nuclear power plant, although a sensitivity study showed that the largest increase in CDF caused by deregulation is 1.5E-5 per reactor-year.

As part of the task action plan, NRR performed a study to assess the potential effect of deregulation on nuclear power plant CDF (Ref. 16). This study modeled an example PWR and boiling-water reactor with baseline SBO CDF of 3.6E-6/reactor-year and 5.3E-6/reactor-year, respectively. The study assumed the values of grid-related LOOP frequency and non-recovery times reported in NUREG-1032, and used the staff's simplified probabilistic assessment risk models (which are used for the ASP program). Sensitivity studies were performed to determine the combination of factors (LOOP frequency, recovery time) needed to increase the baseline CDFs to the SBO goal of 1E-5/reactor-year. For the example PWR, the grid-related LOOP frequency must increase by more than a factor of 10 (from 0.01/reactor-year to more than 0.1/reactor-year) and the expected offsite recovery time must double before the SBO goal is compromised. For boiling-water reactors, the grid-related LOOP frequency must increase by about six times (from 0.01/reactor-year to 0.06/reactor-year) and the expected offsite recovery time must double before the SBO goal is compromised. Other combinations of LOOP frequency and recovery time to meet the SBO goal are given in the NRR report.

3.3.6 Onsite Alternating Current Emergency Power System Reliability

In consideration of the potential impact of deregulation on a nuclear plant SBO, it is equally important to review the functional reliability of the emergency power system used to mitigate the LOOP and prevent a LOOP event from progressing to an SBO event. The Idaho National Engineering Laboratory (INEL) (now INEEL), under NRC contract, completed a reliability study, INEL-95/0035, "Emergency Diesel Generator Power System Reliability, 1987–1993," February 1996 (Ref. 17). INEL-95/0035 indicated that the reliability estimate for 29 plants reporting under Regulatory Guide (RG) 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric power Systems at Nuclear Power Plants" (Ref. 18), with a 0.950 target goal is 0.987. The target reliability estimate for the emergency diesel generators (EDGs) at RG 1.108 plants with a 0.975 target goal is 0.985. This study notes that the target reliability goals do not require or account for unavailability of the EDG due to maintenance out of service. This study shows maintenance out of service could contribute to EDG reliability.

A letter issuing the report noted that "The overall nature of the failures experienced by the plants reporting per RG 1.108 during actual demands differed somewhat from those discovered during monthly surveillance testing, engineering and design reviews, and routine inspections. This indicates that the current testing and inspection activities may not be focusing on the dominant contributors to unreliability during actual demands and may need to be modified to better factor in the conditions and experiences gained from actual system demands." INEL plans to complete an update of the INEL-95/0035 study by January 2000.

3.3.7 Summary Evaluation and Recommendations for Sections 3.3.1 through 3.3.6

Risk evaluations performed by the staff indicate that there is margin to accommodate the potential increase in risk due to deregulation or that the potential increase in the risk resulting from deregulation is likely to be low. In addition, the risk margins do not provide justification for either licensee actions or vigilant NRC action as recommended in ORNL/NRC/LTR/98-12. As stated in the introduction to this report, IN 98-07 was issued to alert licensees to the potential effects of deregulation on the reliability of the offsite power source. Consequently, the staff should take no further regulatory action.

The need for monitoring and follow-up on selected events by the staff is supported by (1) NERC and ORNL reports that indicate that the impact of deregulation may result in regional grid-reliability concerns, (2) some cases in both the NERC and the ORNL/NRC reports that indicate common regional grid reliability concerns, (3) the RES analysis that indicates that individual plants may possibly exceed the SBO objective of 1E-5 per reactor-year, (4) an increasing trend in the grid-related recovery time from 36 minutes to 140 minutes, (5) all grid-related LOOP events meeting or exceeding the ASP threshold of

1E-6 CCDP, and (6) staff observations in INEL-95/0035 about EDG reliability. The staff should

- Monitor LOOP events to detect (and ensure correction of) increases in grid-related LOOP frequency or duration before they become a significant contributor to risk from LOOP events. All grid-related events; plant-centered, grid-initiated events; and events of national interest that affected a nuclear plant and are reported in accordance with 10 CFR Part 50.72 or 10 CFR 50.73 should be considered for evaluation as follows: For events that meet or exceed the ASP CCDP of 1E-6 or have a duration in excess of 30 minutes, onsite and grid control center follow-up, such as an augmented team inspection, should be considered, using the protocol in ORNL/NRC/LTR/98-12, updated to specifically address ORNL concerns, as a guideline.

- Take a forward look at grid reliability by reviewing NERC reliability assessments. NERC grid-reliability concerns should be reviewed by the staff and discussed with NERC as appropriate. In addition, communication with NERC, EPRI, and FERC at the program level, should enhance the forward-looking view of deregulation, including ongoing programs and potential weaknesses.

- The known causes of diesel generator unreliability identified in INEL-95/0035 should be investigated. In addition, the staff should ensure that the reliability of the onsite diesel generators, to include maintenance out of service, is maintained commensurate with the risk studies used to develop the SBO rule.

4 CONCLUSION

Evaluations performed by the staff indicate that the potential increase in risk resulting from grid-related LOOP events due to deregulation is likely to be low; however, the staff will continue to monitor grid reliability and take action, as needed. For example, the NERC reliability assessments and site visits indicate common grid reliability concerns. While the NRC does not have jurisdiction over operation of the grid, Information Notice 98-07, "Offsite Power Reliability Challenges From Industry Deregulation," February 27, 1998," alerted licensees to the potentially adverse effects of deregulation of the electric power industry on the reliability of the offsite power source. Consequently, nuclear power plants are expected to prepare for these concerns by ensuring that plant features for coping with LOOP and SBO events are properly monitored and maintained. In addition to the appropriate command, control, and communication infrastructure with the grid-controlling entity, existing regulatory controls should ensure the reliability of emergency power generators and the adequacy of protective relays and alarms for the switchyard and emergency buses.

The NRC will continue to promptly assess LOOP events as part of the inspection program and also as part of the ASP program. For events that exceed the ASP threshold of 1E-6, further review will be performed, where appropriate, to obtain plant-specific and potential generic insights concerning the event. If the inspection or ASP program reviews indicate that additional staff evaluation of the event is needed, the status of the plant response to deregulation concerns will be assessed using as a guide the protocol developed by ORNL for the site visits. This information will indicate if more plant-specific or generic attention is necessary.

In addition, review of the NERC grid-reliability forecasts and follow-up discussions, as required, appear to be the most practical means of assessing the potential impact of deregulation on the offsite power system. Continued contact with NERC, FERC, and EPRI will also enhance the NRC's understanding of potential deregulation issues related to grid reliability.

5 RECOMMENDATIONS

On the basis of the staff's evaluation of the initiatives completed to date, the following recommendations were developed and subsequently noted in SECY 99-129 (Ref. 8).

(1) The staff will take no further regulatory action to address grid reliability associated with the deregulation issue.

(2) To ensure that the licensing basis is maintained, the staff will follow up on the NERC and site visit concerns, risk-informed analyses, operating experience, and ASP evaluations as follows:

(a) The staff will evaluate the adequacy of (i) the existing technical guidance on offsite power and voltage issues, (ii) the degraded voltage protective relay setpoints, and (iii) the scope of the offsite power system frequency protection, including whether the existing reactor coolant pump underfrequency protection could lead to unnecessary trips. These actions will ensure that plant ac safety equipment remains protected from abnormal offsite system voltages and frequencies.

(b) The staff will investigate causes of diesel generator unreliability identified from INEL-95/0035, "Emergency Diesel Generator Power System Reliability 1987–1993," February 1996. The staff will continue to assess the reliability of the onsite diesel generators to ensure that the reliability is maintained consistent with the risk studies used to develop the SBO rule (10 CFR 50.63).

(c) The staff will continue to assess significant LOOP events that are reported in accordance with 10 CFR 50.72 and 50.73, for prompt review as part of the inspection program. The 10 CFR 50.73 LOOP events will also continue to be reviewed as part of the ASP program. Follow-up action will be considered, as indicated by the inspection program, for LOOP events that either meet or exceed the ASP conditional core damage probability of 1E-6, or that last longer than the national median time of approximately 30 minutes.

(d) The staff will remain cognizant of the current status of grid issues, and will assess future electric power grid reliability and its potential impact on nuclear power plants' offsite power systems through its continued contacts with NERC, the Federal Energy Regulatory Commission and the Electric Power Research Institute.

6 REFERENCES

1. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, "Briefing on Electric Grid Reliability, April 23, 1997, and Briefing on Electric Utility Restructuring, April 24, 1997," May 27, 1997.

2. U.S. Nuclear Regulatory Commission, Information Notice 98-07, "Offsite Power Reliability Challenges From Industry Deregulation," February 27, 1998.

3. U.S. Nuclear Regulatory Commission, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996," NUREG/CR-5496, June 1998.

4. U.S. Nuclear Regulatory Commission, "Evaluation of Station Blackout Accidents at Nuclear Power Plants," NUREG-1032, June 1988.

5. U.S. Nuclear Regulatory Commission, "Regulatory/Backfit Analysis for the Resolution of Unresolved Safety Issue A-44, 'Station Blackout'," NUREG-1109, June 1988.

6. U.S. Nuclear Regulatory Commission, "Evaluation of the Reliability of the Offsite Power Supply as a Contributor to Risk of Nuclear Plants," ORNL/NRC/LTR/98-12, August 1998.

7. North American Electric Reliability Council, "Reliability Assessment, 1997–2006," October 1997.

8. North American Electric Reliability Council, "Reliability Assessment, 1998–2007," May 1998.

9. Electric Power Research Institute, "Breakthrough in Stability Assessment," *EPRI Journal*, August 1996.

10. U.S. Nuclear Regulatory Commission, letter to All Power Reactor Licensees, "Adequacy of Station Electric Distribution System Voltages," August 8, 1979.

11. U.S. Nuclear Regulatory Commission, SECY 99-129, "Effects of Electric Power Industry Deregulation on Electric Grid Reliability and Reactor Safety," May 11, 1999.

12. Western System Coordinating Council, "WSCC System Disturbance Report," December 14, 1994.

13. Western System Coordinating Council, "WSCC Preliminary System Disturbance Report," December 8, 1996.

14. U.S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, letter from C.E. Rossi, "Issue of Special Report–Emergency Diesel Generator Power System Reliability 1987–1993, INEL-95-0035," March 22, 1996.

15. Lee Abramson, U.S. Nuclear Regulatory Commission, Probabilistic Risk Analysis Branch, Division of Risk Analysis and Application, Office of Nuclear Regulatory Research, "Risk Significance of Potential Grid Unreliability Due to Deregulation," June 1999.

16. Nicholas T. Saltos and Daniel M. O'Neal, "Risk Impact of Electric Grid 'Deregulation' on Nuclear Power Plants," September 2, 1998.

17. U.S. Nuclear Regulatory Commission, "Emergency Diesel Generator Power System Reliability 1987–1993," INEL-95/0035, February 1996.

18. U.S. Nuclear Regulatory Commission, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Regulatory Guide 1.108.

The New NRC Generic Issue Resolution Process including Technical Results from GI-158, "Performance of Safety-Related Power Operated Valves Under Design Basis Conditions," and GI-165, "Spring-Actuated Safety and Relief Valve Reliability."

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Abstract

The Generic Issues Program first began formally in response to a Commission directive in October of 1976. In 1983, it became one of the first programs to make successful use of probabilistic risk information to aid in regulatory decision making. In the 16 years since the program became quantitative, 836 issues have been processed. Of these, 106 reactor safety issues were prioritized as requiring further evaluation to determine the final resolution. Approximately a dozen generic issues remain unresolved. Although there is far less reactor licensing activity than in the 1970s, new issues continue to be identified from research and operational experience. These issues often involve complex and controversial questions of safety and regulation, and an efficient and effective means of addressing these issues is essential for regulatory effectiveness. Issues which involve a significant safety question require swift, effective, enforceable, and cost-effective regulatory actions. Issues that are of little safety significance must be quickly shown to be so and dismissed in an expeditious manner so as to avoid unnecessary expenditure of limited resources and to reduce regulatory uncertainty. Additionally, in the time since the generic issue program began, probabilistic risk assessment techniques have advanced significantly while agency resources have continued to diminish. Accordingly, the paper discusses the steps that have been taken to enhance the effectiveness and efficiency of the generic issue resolution process. Additionally, two recently-resolved issues are discussed, along with key elements of a proposed new procedure for resolving potential generic issues.

Introduction

A generic issue is defined as a concern that is applicable to all, several, or a class of nuclear reactors or reactor-related facilities. Generic issues have been classified into several distinct categories:

- A Generic Safety Issue (GSI) is defined specifically as a safety concern that may affect the design, construction, operation, or decommissioning of all, several, or a class of reactors or facilities and may have the potential to require licensees to make safety improvements and/or require the promulgation of new or revised requirements or guidance.

- A Regulatory Impact Issue (RI) is not related to improving safety but to modifying current NRC requirements or guidance with the primary purpose of reducing the regulatory impact, usually cost, of requirements on licensees or applicants.

- An Environmental Issue (EI) involves impacts on those items protected by the National Environmental Policy Act (NEPA).

- A Licensing Issue (LI) addresses actions the NRC staff should take to (a) increase its knowledge, certainty, and/or understanding of safety issues in order to gain confidence in assessing levels of safety; (b) improve or maintain the NRC capability to make independent assessments of safety; (c) establish, revise, and carry out programs to identify and resolve safety issues; (d) document, clarify, or correct current requirements and guidance; or (e) improve the effectiveness or efficiency of the review of applications.

- An Unresolved Safety Issue (USI) is related to generic issues and is defined as a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed and that involves conditions not likely to be acceptable over the lifetime of the plants affected. USIs can be thought of as exceptionally important GSIs. The two types are treated identically, except that the progress on any issue defined as a USI must be regularly reported to Congress.

There are two major documents which govern the NRC staff's procedure for dealing with generic issues: NUREG-0933, "A Prioritization of Generic Issues," which is updated semiannually and is publicly available on the NRC web page, and NRC/RES Office Letter 7, "Procedures for Identification, Prioritization, Resolution, and Tracking of Generic Issues."

History of Program

The generic issues program began in the 1970s, well before the TMI-2 accident. Reactor licensing was extremely active because of the number of applications and because of the large number of individual plant designs under review. Moreover, the regulations (e.g., Appendix K to 10 CFR 50) were being revised and updated. The Standard Review Plan was just being issued, with the purpose of bringing both uniformity and efficiency to the review process.

As this activity progressed, a number of issues arose which were applicable to more than one docket. Rather than resolve these issues on individual dockets, the generic issue program was begun, with the purpose of providing uniform resolutions, and also saving the administrative effort required for consideration of each issue as an "open item" on each docket. (Most of the licensing activity was on construction permits, and it was assumed that the various issues would be resolved before issuance of an operating license.)

The number of issues became quite large, and it was necessary to set some priorities to make the best use of the agency's limited staff resources. Three schemes were used:

- In 1977, all existing issues were classified into four categories according to importance, from "significant" to "little or no importance." This classification was done by a review committee, using a point scale. It is this classification which resulted in the letter designations found in the older generic issues, e.g., A-45, "Decay Heat Removal," and B-17, "Criteria for Safety-Related Operator Actions." Some of the "A" category issues were designated as Unresolved Safety Issues.

- In early 1978, the issues were reclassified into Groups 1 through 8, this time by type of issue rather than by order of importance.

- Late in 1978, the staff began using risk assessment to place the issues into four categories ranging from Group I (potential high risk) to Group IV (items not directly related to risk). This classification was not quantitative in nature. Instead, a group of people who had been involved in the WASH-1400 project[1] was convened. This group used its extensive experience and knowledge of which systems and event sequences were important to risk to place each generic issue into an appropriate category.

At this point, there were 142 generic issues in the system, which made for rather extensive discussions in the committee meetings associated with the classifications listed above. The TMI-2 accident added approximately 400 more generic issues. It was at this point that a new, more quantitative prioritization approach was tried. This was done in two stages:

- In 1981-82, a PRA-based quantitative methodology was developed. Issues were evaluated, usually using rudimentary event tree calculations, and a risk figure was developed for each issue. The objective was to develop an ordered list.

- In 1983, this methodology was redefined to conform to the Safety Goal Policy Statement, which was under development at the same time.

The quantitative methodology had several major advantages:

- Even though the calculations were quite basic, the quantitative approach forced the analyst to thoroughly research the safety significance of the issue in a disciplined manner.

- When quantifying an issue, it became very easy to separate out regulatory impact, environmental, and licensing issues.

- The calculations and the associated writeups provided a thorough, reasonably comprehensible explanation of how the priority figure was generated. This written record could then be reviewed by all interested parties. If omissions or errors were found, the calculation could be corrected.

- The prioritizations could be done by a large number of analysts working independently in parallel, as opposed to panels or committees which evaluated issues sequentially.

- Once the method was renormalized to incorporate the Safety Goal Policy Statement, a low score became a legitimate, defensible reason to drop an issue entirely, rather than to just assign a low priority. Issues of little or no safety significance could be dropped from any further consideration at the prioritization stage.

Management responsibility for the generic issues program was moved from the Office of Nuclear Reactor Regulation to the Office of Nuclear Regulatory Research in 1987. In April 1993, after approximately ten years of experience with the methodology, adjustments were made in the numerical thresholds for categorizing generic issues, while retaining the basic features of the method. These adjustments involved raising risk thresholds and simplifying the way in which costs entered the priority rankings. What motivated the raising of risk thresholds was the observation that, of the issues resolved, only 3 of the 27 medium-priority issues and about half of the high-priority issues resulted in decisions to take regulatory action, i.e., in retrospect, it appeared that resources had been devoted to resolving a large number of issues with no resultant safety improvement. (This outcome must be interpreted with the qualification that generic issue resolution efforts that have not led to regulatory action have, nevertheless, in many instances, produced safety benefits through licensee actions taken voluntarily, in consideration of the issues raised, or in response to interim guidance.) However, the extent of these benefits, when they occurred, was generally in proportion to the priority rank, and medium-priority issues usually resulted in marginal improvements. The proposed revisions were submitted to the Commission in SECY-93-108, "Revised Guidelines for Prioritization of Generic Safety Issues";[2] in July 1993, Commission approval was obtained.

The threshold adjustments were intended to cause the prioritization process to model the resolution process without the earlier, apparently excessive margin for initial uncertainties, in order to reduce resources expended on analysis, evaluation and review efforts that do not produce safety improvements, while still ensuring attention to issues that require it. The raising of the numerical safety thresholds was accompanied by strengthened attention to uncertainties and special considerations, to help recognize instances when a priority rank higher than the indication from the new numerical formula was warranted, the objective being to improve the efficiency of the prioritizations without impairing their validity.

Generic Issue Process

Generic issues have been addressed using six separate and distinct steps: identification, prioritization, resolution, imposition, implementation, and verification. An explanation of each of these six steps is given below (see Figure 1).

Identification. Generic concerns may be identified by individuals or organizations within the NRC staff or by the Advisory Committee on Reactor Safeguards (ACRS), the nuclear power industry, or the public. RES Office Letter No. 1, "Procedures for Identification, Prioritization, Resolution, and Tracking of Generic Issues,"[3] provides a procedure and suggested content for individuals or organizational units within the NRC to request consideration of a concern as a new generic issue. This procedure may also be used by parties outside the NRC to express their concerns to the staff for consideration as potential generic issues. Sources of potential generic issues are many and varied and include, but are not limited to, the following: evaluation of safety-related research, operating experience reviews, risk assessment analyses, and public and industry concerns.

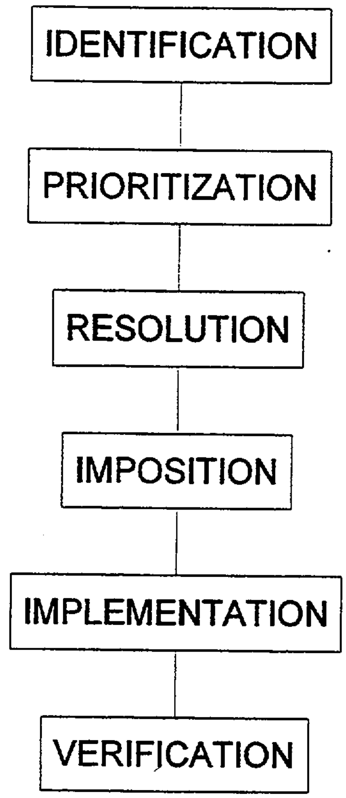

Figure 1. Current Generic Issue Process (from NUREG-0933)

Prioritization. The prioritization step is described in detail in the introductory chapter of NUREG-0933 "A Prioritization of Generic Safety Issues." [4] Estimates are made of the change in core damage frequency, public risk (as measured by the potential change in total whole-body person-rem out to a radius of 50 miles), total cost of resolution, and remaining plant lifetime. The resulting parameters (ΔCDF, public risk, and value/impact ratio) are compared with the priority decision criteria in NUREG-0933 to determine a priority rating. Generally, the prioritization write-up is circulated to the person or organization that raised the issue, the organizational unit that would eventually have responsibility for resolving the issue, and some persons with expertise in PRA and in the technical area involved in the issue. Comments from all of these entities, which often have very different viewpoints, are incorporated before the prioritization is issued in final form.

Many issues are dropped at the end of the prioritization stage, because of little or no safety significance. Because this is the end of the process for such issues, some conservatism is built into the process to ensure that marginal issues are not dropped inappropriately. It is a truism among analysts that one must be much more careful and thorough when dropping an issue, and the write-ups are often more detailed for such issues.

Prioritization calculations are fairly limited and are performed using what information is readily available. Once an issue receives a high or medium priority, a task action plan is written, and it competes for resources just as would any other agency project. The obtaining of more or better information, by means of calculations, experiments, or requests of the industry, is done at the resolution stage.

Resolution. After an issue has been prioritized and approved for resolution (i.e., not dropped following prioritization), the first task is the development of a plan to delineate the work to be done, assignment of major responsibilities, identification of project resource needs, and scheduling of milestone dates. These activities vary in scope and depth in accordance with issue priority and the depth of information on a given issue. The second task involves development of a technical solution. Typically, the information used to resolve an issue comes from experience data, experiments, tests, analyses, and probabilistic risk assessments.

In the final stage of resolution, the technical findings are used as a basis to develop a proposed resolution for the issue involving a change to NRC requirements or guidance. Several alternatives may be considered. A regulatory analysis, including a detailed cost/benefit analysis of each practical alternative, and consideration of the best methods of imposition, implementation, and verification are used in selecting a proposed resolution. One alternative may be the imposition of a "backfit," as defined in Part 50.109 of Title 10 of the Code of Federal Regulations. If a backfit is proposed, first, a determination is made as to whether the backfit is in fact required to provide adequate protection to the health and safety of the public or simply provides for enhancement of public health and safety. If it is determined that the backfit is necessary to provide an adequate level of protection, the backfit will be imposed regardless of the costs to achieve it. If it is determined that the backfit provides for enhancement of public health and safety, a generic analysis is required that assesses the nine factors specified in 10 CFR 50.109(c). Once the cognizant NRC Office Directors have agreed to a proposed resolution, it is then forwarded to the Committee for the Review of Generic Requirements (CRGR), the ACRS, the Executive Director for Operations (EDO), and the Commission for review and approval as appropriate. Changes to regulations, Policies, the Standard Review Plan (SRP), and Regulatory Guides are published in the Federal Register for public comment. Comments received are then incorporated, as appropriate, with the final product published in the Federal Register. Resolution of a generic issue can take from several months to a few years depending on the length of time required by the deliberations involved at each of the above steps.

Imposition. Imposition is the step in the generic issues program where each affected licensee and/or applicant is required or guided to prepare a schedule for implementing the generic issue resolution consistent with a Rule, Policy, Regulatory Guide, generic letter, bulletin, and/or licensing guidance developed during the resolution stage

For backfits, imposition is considered complete when each affected licensee has committed to compliance actions and schedules for implementing these actions. For forward-fits, i.e., changes in NRC requirements or guidance that are put in place before an applicant submits an application, the imposition of a generic issue resolution is complete when the new requirement or guidance becomes effective as an integral part of NRC regulations, policies, and/or guidance.

Implementation. Implementation is the step in the generic issues process where the affected licensees perform the actions on existing plants to satisfy the commitments made during the imposition stage. These may include modifications/additions to equipment, structures, procedures, technical specifications, operating instructions, etc. No later than 30 days after each affected licensee has completed all of the actions required for a particular generic issue resolution, and the modified/additional system is fully operational, the licensee is required to certify in writing to the NRC that plant modifications/additions have been completed in accordance with NRC requirements, policies, and/or guidance. When all affected licensees have officially notified the NRC of completion of all required/committed actions, the implementation stage is complete, unless it is determined by the staff from subsequent verification inspection that additional licensee actions are needed for compliance.

Verification. The verification step consists of three parts. First, the portions of a licensee's actions, if any, that warrant NRC inspection must be determined. This decision is made during the resolution stage based on the judgment of the safety significance of the issue relative to other matters in the inspection program, licensee performance, and the resources needed to accomplish a meaningful inspection. Next, as necessary, inspection instructions are prepared to ensure that the NRC inspection is performed in a consistent and appropriate manner at all affected plants; the inspection, by its very nature, is an audit. Therefore, carefully thought-out instructions must be developed and provided to the NRC inspectors so that the maximum safety benefit is achieved for the limited resources devoted to this effort. The third part of the verification process is the actual verification and documentation of the results in an inspection report. Physical inspections are performed on an audit basis in a manner consistent with general inspection procedures which involve a sampling of changes made by licensees or applicants, as opposed to a 100% inspection of all actions.

Recent Generic Issues

Two recently-resolved generic issues are illustrative of the current process. Both issues were not dropped following the prioritization step, and task action plans were developed and carried out for these issues.

Generic Issue 158, "Performance of Safety-Related Power-Operated Valves Under Design Basis Conditions," was first identified in 1991. The issue was initiated because reactor operating experience and research results on solenoid-operated, air-operated, and hydraulically-operated valves indicated that testing under static conditions did not always reveal how these valves would perform under design basis conditions. A number of failures had occurred because of inadequate design, installation, or maintenance. (A similar concern on motor-operated valves was investigated under GI-II.E.6, and eventually resulted in the issuance of Generic Letter 89-10.)

A prioritization calculation was performed by changing the failure probabilities for these valves in several PRAs, and observing the change in core damage frequency. Based on these calculations, a priority of "medium" was assigned to the issue, and a task action plan was generated to resolve it.

After considerable investigation in the resolution step, it was discovered that, although there is safety significance for this issue, there is no cost-effective generic solution - the use of these valves, and consequently the nature of any program to increase their reliability, varies greatly from plant to plant. However, the lack of a generic solution did not imply that nothing should be done, and several programs have been voluntarily undertaken by industry groups to increase the reliability of these valves. The issue was judged to be a "compliance" issue as defined in 10 CFR 50.109 paragraph (a)(4)(i), and removed from the Generic Issues program. Instead, the NRC staff is monitoring industry voluntary efforts to develop an acceptable guidance document. If licensees do not take adequate action to address questions related to power-operated valve function under dynamic design basis conditions, appropriate regulatory action will be taken.

Generic Issue 165, "Spring-Actuated Safety and Relief Valve Reliability," was first identified in 1992. This issue was identified by NRR when it was found that, on a number of occasions, licensees reported that spring-actuated safety valves and relief valves failed to meet setpoint criteria within the desired tolerance. Other reported incidents included more seriously degraded performance of safety and relief valves. These results suggested that other systems with safety and relief valves could be adversely affected by setpoint drift. More importantly, at Shearon Harris, the failure (open) of a high head safety injection system relief valve at a very low setpoint resulted in the undetected degradation of the entire system and would have resulted in inadequate emergency core coolant injection if a small-break or intermediate-break LOCA had occurred.

Spring-actuated safety valves and relief valves provide overpressure protection for a number of fluid systems in both PWRs and BWRs. However, the spurious opening failure of these valves in safety-related support systems could cause a significant diversion of flow from these systems and thus prevent the systems from performing their designed function. A prioritization calculation using this assumption resulted in this issue being given a "high" priority designation

A task action plan was generated and approved, and a full investigation of the issue was performed. In the course of this investigation, it was discovered that, although many systems are equipped with small safety or relief valves, these valves are virtually always too small to divert sufficient flow to fail the system. Only in a few, very rare instances was there any possibility of causing a safety system failure even if one of these valves failed in the full open position. Those few instances where an "oversized" valve could have caused system failure by diverting flow were investigated and either dismissed on probabilistic grounds or resolved by voluntary action on the part of the licensee. The issue was declared "resolved" and closed.

Planned improvements with the New Generic Issue Resolution process

Although the current generic issue process has been used successfully for many years, a number of changes have been suggested and included in Draft Management Directive 6.4, "Generic Issue Program." The new process will be used to address generic issues proposed after August 1, 1999.

– Generic <u>Safety</u> Issues will be classified as "adequate protection," "substantial safety enhancement," or "burden reduction" issues, for consistency with 10 CFR 50.109, the "backfit rule." (These terms did not appear in 50.109 until approximately 1988.)

- Administrative and licensing issues will be addressed in other programs, and will no longer be addressed by the Generic Issue Program.

The workflow steps will be modified as follows (see Table 1):

Table 1: Steps in Current and New Generic Issue Processes

Current Generic Issue Process Step	New Draft Management Directive 6.4, "Generic Issues Program" Process Step
Identification	Identification
Prioritization	Initial Screening
	Technical Screening
Resolution	Technical Assessment
	Regulation and Guidance Development
Imposition	Regulation and Guidance Issuance
Implementation	Implementation
Verification	Verification

As can be seen from Table 1, several steps will be modified:

- The prioritization step will be dropped and replaced with two steps involving an initial screening step (performed by review panels), to remove compliance issues and the like, and a technical screening step, similar to the quantitative prioritization in the current process. In view of the fact that the large backlog of issues of the 1970s and early 1980s has been worked off, and only a small number of new issues are now expected each year, it is no longer necessary to give a priority. Instead, the technical screening will only result in a conclusion of "continue" or "drop."

- The "Resolution" step will be divided into a "Technical Assessment" step (which will provide the regulatory analysis necessary to proceed) and "Regulation and Guidance Development" step. In the current process a generic issue is considered "resolved" once the agency decides upon a course of action. The terms "resolution" and "resolved" were often confused with another term, "closure." To clarify terminology, the new process defines an issue as "resolved" once all steps have been completed, including "Implementation" and "Verification."

- The "Imposition" step will become the "Regulation and Guidance Issuance" step. In addition to the name change, process guidance is now provided.

- "Implementation" and "Verification" steps will have the same titles. However, process guidance is now provided.

Technical improvements

In addition to the procedural changes described above, several technical changes are planned. When quantitative techniques were first introduced into the generic issue program in the early 1980s, most of the calculations were done using WASH-1400 techniques and data. When personal computers became available, many calculations were based on the Reactor Safety Study Methods Application Program (RSSMAP) studies for the Oconee and Grand Gulf plants. Some of these techniques are still in use today. Several improvements are planned:

First, newer PRA models will be used for those calculations where ΔCDF and change in public risk are estimated by varying the parameters in an existing PRA. The RSSMAP PRAs are obsolete both in calculational techniques and in modeling (e.g., reactor coolant pump seal leaks are not modeled). In addition, the two plants that were used, Grand Gulf and Oconee, are not very representative of the spectrum of plants currently operating.

The staff is beginning to use the Rev. 2QA SPAR models[5] developed for the Accident Sequence Precursor program. This is a set of models which cover virtually every domestic operating plant. Moreover, these models are already loaded into the SAPHIRE code package[6].

There are limitations to the SPAR models - not all support systems are modeled, and, because they were intended for the analysis of precursor events, large break LOCA sequences are not included. When the Rev. 3 SPAR models are issued, both of these limitations will be addressed. Moreover, the Rev. 3 models will include uncertainties, which will allow the use of uncertainty analyses in generic issue calculations for the first time.

When a more detailed model is needed, or when external events must be considered, the PRA models in NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants,"[7] can be used. However, there will always be cases where a new model must be constructed - no existing PRA can include all possible phenomena, systems interactions, etc., which may be present in a generic issue.

Second, the consequence analysis will be updated. These calculations are still based on calculations of WASH-1400 release categories. As resources become available, new consequence analysis tables will be generated.

Conclusion

The current generic issue process has worked well for many years. To assess the program's regulatory effectiveness (both the current process and the new process), the following should be considered:

- *The process is directly related to public health and safety.* From its inception, risk has been the yardstick by which the most important (i.e., safety-significant) issues were worked on first. The risk yardstick has also been used to eliminate issues with little or no risk significance, and, in accordance with the backfit rule (10 CFR 50.109), risk and cost effectiveness are the final measures which determine what regulatory action is taken.

- *The process makes maximum use of risk information.* This program (along with the Accident Sequence Precursor program) was one of the first to make use of probabilistic risk assessment techniques in regulatory decision making.

- *The methods and criteria are objective, clear, and public.* The criteria are those of the Safety Goal Policy Statement. The methods have long been published in NUREG-0933, which is now available on the agency's World Wide Web site as well as being available in printed form. Updates are published every six months.

- *The treatment of each individual issue is open, defensible, and public.* The individual write-ups on each issue are also published, in both printed and electronic form, in NUREG-0933. The treatment of each issue is reviewed by both the originator of the issue, and by the entity which will have responsibility for any regulatory action. These two may be the same entity, but more often have somewhat of an adversarial relationship where the specific generic issue is concerned. The treatment of the issue is defended at both the prioritization and resolution stages.

It is therefore concluded that this program does contribute significantly to regulatory effectiveness.

References

1. WASH-1400 (NUREG-75/014), "Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," U.S. Atomic Energy Commission, October 1975.

2. SECY-93-108, "Revised Guidelines for Prioritization of Generic Safety Issues," April 28, 1993.

3. Memorandum to all RES Employees from David L. Morrison, Director, Office of Nuclear Regulatory Research, "RES Office Letter No. 7--Procedures for Identification, Prioritization, Resolution, and Tracking of Generic Issues," dated February 16, 1996.

4. NUREG-0933, "A Prioritization of Generic Safety Issues," December 1983, and Supplements 1 to 19.

5. S. M. Long, P. D. O'Reilly, E. G. Rodrick, and M. B. Sattison, "Current Status of the SAPHIRE Models for ASP Evaluations," presented at the Probabilistic Safety Assessment & Management (PSAM IV) conference, New York, 1998.

6. NUREG/CR-6116, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE)," July 1994.

7. NUREG-1150, "Severe Accident Risks: An Assessment for Five U. S. Nuclear Power Plants," December 1990.

27th Water Reactor Safety Information Meeting

Enhancing Regulatory Effectiveness

Opportunities and Issues in the Robust Fuel, Probabilistic Risk Analysis Standards Development and Graded QA Programs

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Bethesda, MD
October 26, 1999

Robust Fuel Program

- EPRI is conducting a research effort with the utilities and fuel vendors aimed at addressing current fuel performance issues and creating the requirements for fuel for operation in the next millennium
- These stakeholders seek the resolution of current fuel performance issues and the availability of fuel with more operating margin and higher licensed burnup
- Significant resources are presently programmed to support expensive testing to resolve issues raised by the imposition of a Reactivity Insertion Accident definition (PWR Rod Ejection) that is not risk-informed

Regulatory Effectiveness Issue

- **<u>Lack of Consistency</u>** with NRC's PRA Policy Statement distorts the Robust Fuel Program's research plan
- Diverts attention from postulated accidents that may be more risk-significant than the Rod Ejection Accident
- Weakens stakeholders' support for the program
- Are other elements of NRC's Research Plan inconsistent with the Commission's PRA Policy Statement ?

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Probabilistic Risk Analysis Standards Development Programs

- Lack of PRA Standard(s) has slowed implementation of risk-informed applications
- Industry and NRC now recognize need for a complete set of PRA Standards and Peer Certification process
- ASME has drafted an at-power PRA standard (limited to internal events)
- ANS is drafting a Low Power & Shutdown PRA standard (including external events)
- ANS has committed to making the Low Power & Shutdown PRA standard "seamless" with the ASME at-power PRA standard

Regulatory Effectiveness Issue

•Consensus standards derive their authority from both their technical content <u>and</u> the process that creates the standard (the power of the "consent of the governed")

•Beyond technical content, the authority of the standard can be diminished if there is a **<u>Lack of Confidence</u>** in the consensus process

Regulatory Effectiveness Issue

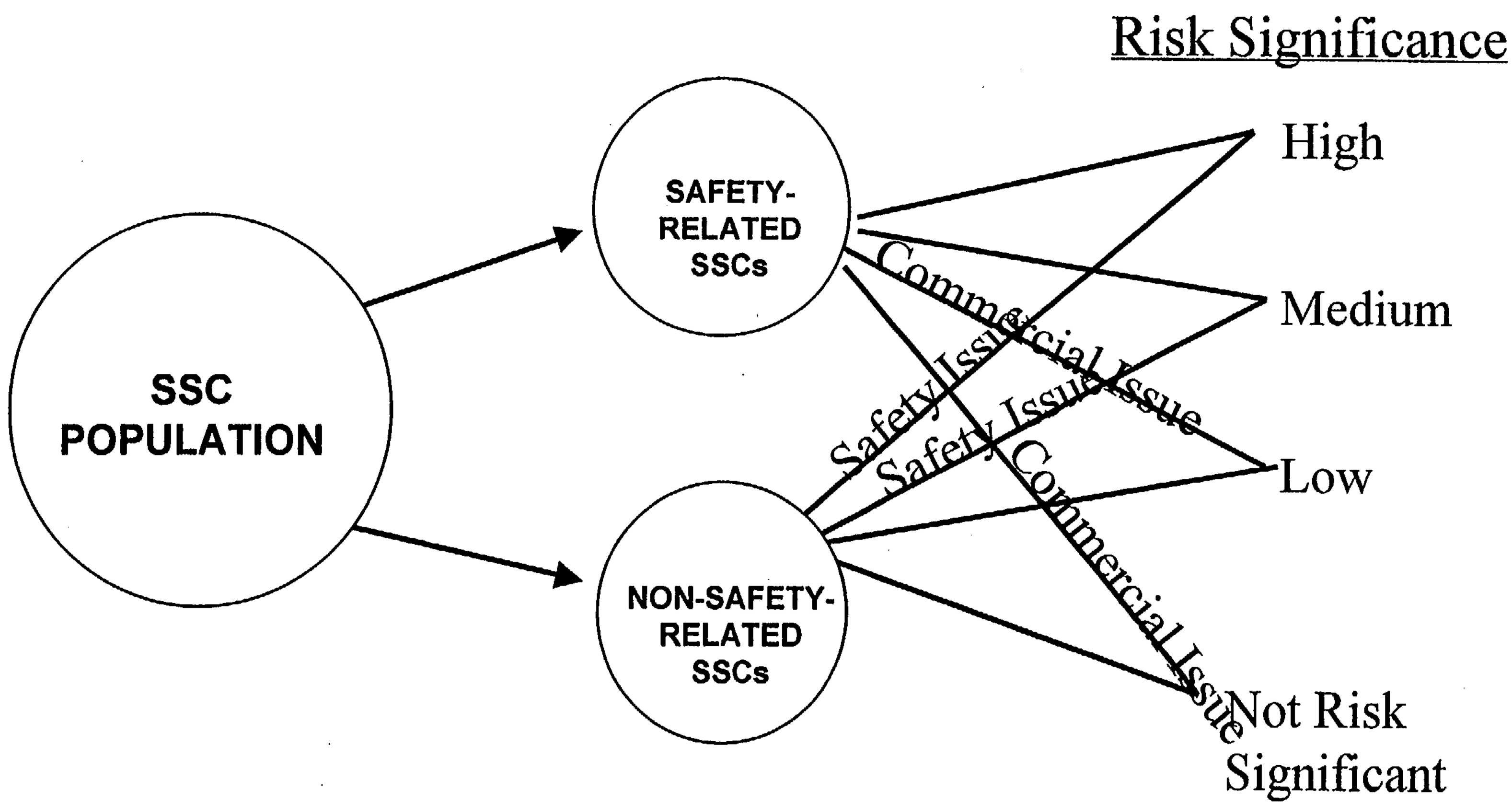
•NRC staff and management (and industry personnel) should participate in the standards development process in ways that foster confidence in both the standards development process and the standard itself

•Successful standards will gain widespread acceptance in the industry, regulatory and insurance communities

Graded QA

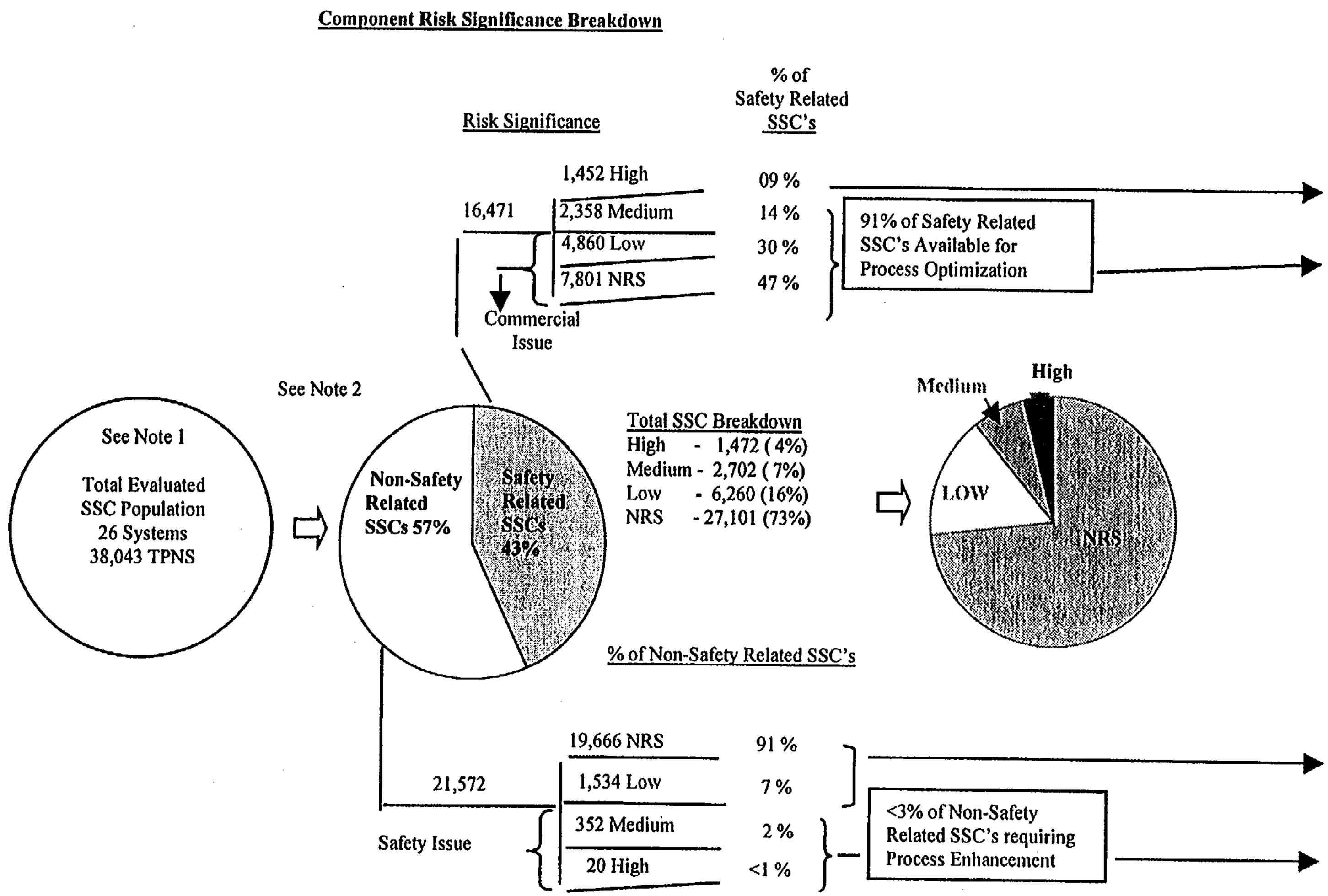
- NRC issued SER on STP Graded QA program in November 1997
- Risk significance evaluation process reviewed in NRC Graded QA SER

QUALITY ASSURANCE
HISTORICAL PERSPECTIVE



SSC = Structures, Systems and Components

COMPONENT RISK SIGNIFICANCE - MAINTENANCE APPLICATION BREAKDOWN
As of 9/22/99 (See Note 1)



Note 1: Inclusive of Systems AF, CC, CH01, CV, DG, DI, DO, DX, EW, FC, FW, HC, HE, HF, HG, HM, IA, JW, LU, MS, RA, RC, RH, SB, SD and SI.

Note 2: Includes Augmented Quality Related 7F, 7S, 7P and 7R

Implementation Issues

- SER did not allow exemption to 10CFR50 special treatment requirements (seismic, EQ, ASME etc.)
- GQA program at STP restricted and anticipated parts procurement benefits are not being realized

Implementation Issues

- STP requested and received guidance from NRC Staff on reclassification of safety-related not-risk significant components to not-safety related (May 28, 1999) using 10CFR50.59

- Guidance is viewed by some at STP as ambiguous

The Path Forward

- To be fully effective, 10CFR50.59 needs to unambiguously allow Risk-Informed judgments to be applied
- Must allow programs, activities, structures, systems and components to be graded commensurate with their importance to nuclear safety

The Path Forward

- SECY 98-300 recommended phased approach to risk-inform 10CFR50 and subsequent SRM authorized NRC staff to proceed

- Previously approved STP Graded QA risk significance determination process anticipates efforts required to risk inform 10CFR50 as recommended by Staff and approved by Commissioners

The Path Forward

- Amend scope to which 10CFR50 applies by redefining safety related and providing selected exemptions to Part 50 (Option 2 of SECY 98-300)

- GQA Risk Significance determination process is an approved method for establishing risk significance and satisfies intent of Option 2

The Path Forward

- Risk inform individual sections of 10CFR50 in next phase (Option 3 of SECY 98-300)

- Lesson Learned from STP experience is that Special Treatment Requirements in 10CFR50 need to be addressed for successful implementation of GQA

Regulatory Effectiveness Issue

- The NRC Staff is grappling on many levels with the issue of the **Lack of Cohesiveness** of NRC regulation.

- The Commission's PRA Policy Statement in August 1995 set the staff and the industry on a course to re-define the regulatory process.

- The efforts to implement such a broad policy initiative have, inevitably, revealed the overlapping regulatory requirements put in place over the years to enhance defense-in-depth and respond to events.

- At present, the NRC staff appears to be working mightily to ***unravel*** the "Gordian Knot"*

- There are other possible strategies.

*Gordian Knot {gohr'-dee-uhn}

In Greek legend, the 'Gordian Knot'was the name given to an intricate knot used by Gordius to secure his oxcart. Gordius, who was a poor peasant, arrived with his wife in a public square of Phrygia in an oxcart. An oracle had informed the populace that their future king would come riding in a wagon. Seeing Gordius, the people made him king. In gratitude, Gordius dedicated his oxcart to Zeus, tying it up with a peculiar knot. An oracle foretold that he who untied the knot would rule over all Asia. According to a later legend, Alexander the Great severed the knot with his sword. From that time, "cutting the Gordian knot"came to mean solving a difficult problem by bold, unorthodox action.

Reducing Burden for Decommissioning Rule Financial Assurance Requirements

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U.S. Nuclear Regulatory Commission

THE PROBLEM:

In 10 CFR 50.75 and 50.82, NRC requires licensees to periodically update the cost estimates for decommissioning their reactors. The estimates are intended to provide reasonable assurance that at any time during the life of the plant, especially with respect to early, premature closure, sufficient money will be available to complete decommissioning. Cost estimates are required to be updated and submitted annually, from the time of initial operations (10 CFR 50.75). These annual estimates must be determined by using a rule-specified formula. The formula includes the cost of waste disposal as well as the costs of labor and energy use. The rule specifies how to evaluate the annual cost contributions for labor and energy using information provided in publications by "the Department of Labor, Bureau of Labor Statistics." The rule directs that NUREG-1307, *Report on Waste Burial Charges*, be used to obtain the information for calculating the cost of waste disposal. NUREG-1307 is updated by the NRC, when necessary, by routinely examining the availability of licensed LLW disposal facilities and associated monetary costs for disposal.

Around 1980, when the rule was originally developed, there was little concern about waste disposal cost becoming a significant contributor to the total cost of decommissioning. It was assumed that cheap, offsite, LLW disposal would continue to be available. Although some consideration was given for using site-specific cost estimate requirements in 10 CFR 50.75, it was ultimately concluded that such a provision would be overly burdensome. There is considerable uncertainty when a decommissioning cost estimate is evaluated early in the life of the reactor. At that time a generic cost estimate, rather than a site-specific one, would also be reasonable to use, but would require less effort to determine. Therefore, it was concluded that an appropriate way to satisfy the cost estimate requirement was to use a generic formula, as specified in the final rule.

From the time the rule became effective (1988), the cost of LLW waste disposal at the licensed disposal facilities has increased significantly. Current evaluations show that the cost of waste disposal has become the dominant cost contributor in many licensees' estimates of total decommissioning cost. When the rule was developed, little consideration was given to minimizing the amount of decommissioning waste generated. Using the current high costs for disposal in the formula results in very high disposal cost estimates, even if the actual volume of waste to be disposed is considerably reduced. The formula does not contain an adjustment factor to account for the cost reduction when the waste volume is reduced.

Industry viewed the increasing LLW disposal costs as a major burden in demonstrating financial assurance. They developed alternatives for managing and disposing of their LLW and developed strategies for effectively reducing this cost. These reductions are achieved through a combination of measures that include: (1) procedures that minimize the generation and package volume of waste, (2) utilization of waste vendors that charge lower rates because of their more efficient handling and treatment of the large amounts of waste they routinely process, and (3) using reduced cost LLW disposal facilities that are licensed to accept only very low activity waste. Unfortunately, the existing regulatory structure did not have the flexibility to permit licensees to incorporate the use of these cost saving activities into their required cost estimate as a means of reducing the financial assurance burden. Licensees raised this issue and NEI requested urgent resolution in two letters to the NRC in 1998. The first letter requested regulatory amendment and the second letter suggested revising NUREG-1307 as a faster alternative to rule amendment.

POSSIBLE RESOLUTION OF PROBLEM:

The NRC staff's independent evaluation agreed with the NEI assessment that rule amendment would not to provide any near term relief. The staff explored ways to use more flexible implementation guidance for the cost estimating requirements prescribed in the rule through revision of NUREG-1307. They concluded that this could be a fast way to provide industry relief, as the NEI had suggested.

Although the NRC supported revising the NUREG, the staff could not ignore the information base used to develop the prescribed cost estimating formula. The waste disposal cost formula component was based on offsite disposal of a fixed amount of waste (approximately 17,000 m^3). Today, this waste volume has now been reduced by about a factor of 2/3 for a PWR, and a factor of 1/3 for a BWR. However, the cost reductions that these waste volume reductions would yield, could not be accommodated using the existing formula. A second industry recommendation suggested that the waste could be shipped, when applicable, to lower rate disposal locations rather than sending the waste to the LLW disposal facilities already included in the current NUREG. LLW is also shipped by licensees to waste processing vendors for disposal. Waste processing vendor disposal costs can be substantially lower that those of the LLW disposal facility. If the waste vendors provided information to the NRC suitable for developing generic disposal rates, then the industry contended that these generic waste vendor rates could be included with the rates of the LLW disposal facilities used in NUREG-1307. This additional information could then be used by licensees to evaluate the waste disposal cost component in the formula.

Subsequently, the NRC obtained the necessary waste vendor information and NUREG-1307 was revised. For flexibility, the licensee could still choose to ship the waste directly to a LLW facility.

Revision of NUREG-1307 was expedited so that an alternative LLW disposal methodology would be available to licensees for use in preparing their 1999 cost update. In less than a month's time, NUREG-1307, Rev. 8 was publically available. This NUREG was published in hard copy and placed on the NRC Web site for quick public access.

CHANGES TO NUREG-1307:

NUREG-1307, Revision 8, now contains two options for estimating the cost of decommissioning waste disposal. The option selected is at the discretion of the licensee. In the first option, the annual cost adjustment for waste disposal is based on the charges a licensee would incur by shipping waste directly to licensed LLW disposal sites. This option is the same as used in past NUREG revisions. In the second option, the annual cost adjustment is based on a new evaluation that uses a generic cost rate for LLW disposal based on the charges a licensee would incur if most of the waste is shipped directly to a waste vendor.

IMPACT ON FINANCIAL ASSURANCE BURDEN AND REGULATORY EFFECTIVENESS:

Use of the waste vendor (second) option can result in a significant reduction in the decommissioning waste estimate. For example, a PWR licensee using the second option can estimate the disposal cost of the decommissioning waste (with a small amount going to Barnwell- which most licensees have as their only alternative) to be about $200 million less than the estimate based on the first option. A rule amendment is still required to fully include the effect of all the cost reducing methods available to licensees (e.g., accounting for the actual reduction of the waste disposed of offsite). The modified implementation now available to the licensees in NUREG-1307 takes account of cost reducing measures that can result in significant reductions in the cost estimate. From the efficiency aspect of regulatory effectiveness, these changes allow more efficient application of industry resources by reducing the amount of funds that need to be set aside to offset decommissioning costs. Regulatory effectiveness also dictates that regulatory decisions should be made without undue delay and the expedited revisions to NUREG-1307 to allow immediate use of the new cost estimates is an example of that.

SYNERGISTIC FAILURE OF BWR INTERNALS

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Boiling Water Reactor (BWR) core shrouds and other reactor internals important to safety are experiencing intergranular stress corrosion cracking (IGSCC). The United States Nuclear Regulatory Commission has followed the problem, and as part of its investigations, contracted with the Idaho National Engineering and Environmental Laboratory to conduct a risk assessment. The overall project objective is to assess the potential consequences and risks associated with the failure of IGSCC-susceptible BWR vessel internals, with specific consideration given to potential cascading and common mode effects. An initial phase has been completed in which background material was gathered and evaluated, and potential accident sequences were identified. A second phase is underway to perform a simplified, quantitative probabilistic risk assessment on a representative high-power BWR/4. Results of the initial study conducted on the jet pumps show that any cascading failures would not result in a significant increase in the core damage frequency. The methodology is currently being extended to other major reactor internals components.

INTRODUCTION

General Design Criteria 2 and 4 require that commercial nuclear reactor structures, systems, and components important to safety be designed to withstand the effects of natural phenomena, such as earthquakes, and the effects of postulated accidents, including loss-of-coolant accidents (LOCAs). Boiling water reactor (BWR) internals components were originally believed to have been designed to accommodate these requirements. However, intergranular stress corrosion cracking (IGSCC) degradation has been observed in both core shrouds as well as a number of other BWR reactor internals components, many of which are important to plant safety.

Although IGSCC of reactor internals had been recognized for over 20 years, this phenomenon received increased attention, beginning when crack indications were reported at core shroud welds located in the beltline region of an overseas BWR in 1990. The core shroud is a stainless steel cylinder that is located inside the reactor vessel. It serves to both provide lateral support to the reactor core and to direct the flow of water inside the reactor vessel, and is generally regarded as a component whose integrity is critical to maintaining core safety. Later, a visual inspection of a U.S. BWR core shroud revealed crack indications at several weld regions. Subsequently, General Electric and the NRC[1,2,3] issued correspondence regarding core shroud cracking.

In addition to the BWR core shroud degradation, other BWR reactor internals components, including shroud support access hole cover welds, jet pump hold-down beams, core spray systems, and top guides have also been experiencing IGSCC degradation over the years.[4] These instances have for the most part been sporadic, were not believed to be of major safety importance, and were addressed by General Electric through notices such as Safety Information Letters (SILs) and by the NRC through Information Notices and a Bulletin that have been issued from time-to-time since about 1980.[5,6,7,8,9,10,11] However, the instances of core shroud cracking served to escalate attention as to the seriousness of the IGSCC problem in BWR reactor internals.

The NRC Office of Nuclear Reactor Regulation has followed the problem and issued a Bulletin, Information Notices, and Safety Evaluation Reports (SERs), as well as a Generic Letter.[2] The primary emphasis has been placed on core shroud degradation, but common mode or cascading failure of other components could also have safety significance. Consequently, this NRC-sponsored program described in this paper has been initiated to conduct a risk assessment investigating the concern of cascading failures of BWR vessel internals.

The objective of the study is to assess the potential consequences associated with the failure of IGSCC-susceptible BWR reactor internals components, both singly and in combination with the failures of others. Specific consideration is given to potential cascading and common mode effects on system performance stemming from cracking of core shrouds and other BWR reactor internals components when subjected to design-basis and beyond-design-basis accident loading conditions such as seismic events.

The focus is on mechanical design, failure locations, consequences, potential accident scenarios, and characterization of risk associated with IGSCC degradation of BWR vessel internals. The scope is limited to the basic risk evaluation, including the following:

- The only degradation mechanism considered in this study is IGSCC, including contributing SCC mechanisms such as irradiation-assisted SCC (IASCC). It is recognized that other degradation mechanisms such as fatigue can act synergistically with IGSCC in that a crack which is initiated by IGSCC can propagate to failure from fatigue.

- The NRC is investigating the causes and contributing aspects of the IGSCC problem in separate programs. Industry groups are investigating inspection, mitigation, repair, or replacement, as well as the causes and contributing aspects of IGSCC.

- There are five currently operating types of BWRs in the U.S., designated BWR/2 through BWR/6. Since there is only a single BWR/1 in operation, Big Rock Point, and it is expected to be decommissioned in the near future, the scope of work in this study does not include the unique BWR/1.

BACKGROUND

The three basic elements that must *all* be present for IGSCC to occur are:

1. a susceptible material
2. a chemically aggressive environment
3. a high tensile stress

Under normal circumstances, the stress must be above the yield stress, which can occur at locations such as residual stresses around welds. However, if certain other factors are present, the conditions for the three basic elements listed above (such as the need for the tensile stress to be above the yield stress), may be somewhat altered.

INITIAL PROGRAM PHASE

The first phase of the study involved acquiring and evaluating relevant background information on IGSCC of BWR reactor internals, to qualitatively assess potential accident scenarios, and to identify scenarios for detailed analyses, that is, those expected to have large effects on Core Damage Frequency (CDF). This phase has been completed.

Accident Sequences

Differences in reactor internals designs and accident mitigation systems for the various BWR types were categorized, the degradation of BWR internals to date was catalogued, and management practices to deal with aging were reviewed. From this background study, various types of *systems* failure modes that could result from simultaneous common mode failures of various combinations of reactor internals were catalogued. This included the consideration of functional losses or significant degradations of certain inside-reactor vessel systems. A similar assessment for various types of *mechanistic* failure modes (i.e., the potential results of physical impacts/interactions, common mode and cascading failures, etc., between various reactor internals components due to failures and/or degradations of the components that are subject to IGSCC degradation), was made. This included the various ways that these components might fail and how those types of failures might affect other components inside the reactor vessel. Approximately 250 different and unique scenarios were identified.

The safety significance was also evaluated. This is generally component specific; however, one common safety significance is a loose part which can result from IGSCC. There are three basic safety consequences:

1. a loose part can inhibit control rod motion
2. a loose part can block or partially block a coolant flow channel
3. a large loose part can impact adjacent components and impair their function

There are other safety consequences not associated with loose parts which could be caused by damage to any of several reactor internals components, such as:

1. increased coolant leakage between plenums
2. damage to emergency coolant or shutdown systems
3. damage to control rods or prevent their motion
4. cause of a reactor coolant system leak

In order to reduce the number of scenarios to be considered, a screening logic based on the safety consequences identified above was prepared. Five criteria were developed that are believed to cover the most important issues necessary to adequately address public safety with regards to reactor vessel internals failures. The screening logic was applied to all 250 initially identified scenarios. Of these, 148 remained after the screening, which reduced the work scope somewhat, but still left a large number of sequences to evaluate. A quantitative risk assessment was subsequently conducted on these remaining 148 sequences, as described in the following section.

Preliminary Qualitative Risk Assessment

A qualitative ranking (based on potential contributions to CDF) was made of potential accident scenarios which can be exacerbated by IGSCC degradation of reactor internals. Various possibilities of single, common mode, and cascading failure sequences were postulated for the high, medium, or low rankings. Although the rankings were qualitative, a NUREG-1150 PRA was used to assist in providing for an estimate for the rankings.

A preliminary risk assessment including a list of potential safety concerns (i.e., possible accident scenarios), deterministically developed, was made. Specific areas were described for each potential accident sequence where additional analyses are needed to provide a more definitive understanding of accident scenarios that involve either simultaneous (i.e., common mode) or cascading (i.e., sequentially caused by other failures). The scope included both deterministic failure considerations and qualitative risk assessments.

Most (about 100) of the scenarios were ranked high. Although there appears to be a large number of scenarios, there is a great deal of redundancy in that the high-ranking scenarios fall into variations of two basic categories:

1. loss of the reactor protection system (RPS)
2. loss of coolant to the core

The high-ranking categories were broken into subcategories to further differentiate between the various causes of loss of the RPS and the various ways that coolant to the core could be lost. The following seven subcategories were chosen:

1. loss of scram capability
2. standby liquid control (SLC) system nonfunctional
3. both RPS and SLC nonfunctional
4. medium LOCA with loss of SLC
5. both high-pressure coolant injection (HPCI) and low-pressure coolant injection LPCI ineffective [no redundant emergency core cooling system (ECCS)]
6. core reflood to two-thirds level cannot be maintained (treated as loss of ECCS)
7. high-pressure coolant system (HPCS) and SLC (through sparger) eliminated (several BWR/5 and BWR/6 plants)

Each of the 100 high-ranked scenarios can be placed into one or more of these subcategories.

CONTINUED PROGRAM PHASE

The initial phase of the project, which was primarily to scope the overall effort, has been completed. The second phase is now underway, in which quantitative calculations will be performed. However, there are many difficulties in carrying out this program, such as:

(a) the large number of components and failure sequences
(b) the different types of BWRs
(c) the difficulty in estimating crack sizes and growth rates
(d) there are a large number of disciplines involved
(e) there are limited "good" PRA and thermal-hydraulic models available

It was decided to narrow the research scope to provide a simplified, cost-effective approach. The following simplifications were proposed as an initial approach:

(a) select a single plant for study
(b) select a single component and probable failure locations for initial calculations
(c) perform minimal calculations and research
(d) develop a methodology to introduce IGSCC-induced failures into an existing PRA which can then be applied to the failure of any BWR vessel internals component
(e) convert an existing TRAC-B model to a representative plant to determine flow characteristics

(f) estimate the failure probabilities and insert events associated with the failure of the selected component into an existing PRA, considering a single failure at the most likely locations, common mode failures, and cascading failure sequences. If successful, then apply to other components
(g) use expert panels to critique methods, offer suggestions for approach, and help in estimating probabilities and uncertainties

Plant Selection
A number of criteria were used to select a plant for study, including:
(a) is there an existing PRA model for the plant (internal and external events)
(b) is there an existing TRAC-B model for the plant (or one for a similar plant)
(c) is the plant typical
(d) older plants were preferred

The Peach Bottom BWR/4 was chosen for study. As a high-power BWR/4, it is the most representative type of BWR. There is a fairly good (but not ideal) PRA, and there exists a BWR/4 TRAC-B thermal-hydraulics model which can be modified to represent the selected plant. Figure 1 shows the general arrangement of a typical BWR/3-BWR/4 plant.

Initial Component Selection
A number of criteria were used to select a specific reactor internals component for study, including:
(a) degradation to date
(b) cascading possibilities
(c) safety significance
(d) typicality

The jet pump was the reactor internals component selected for study, as there has been recently discovered cracking in jet pump riser inlet welds[11] and jet pump failure could lead to a variety of cascading failure sequences. IGSCC failures have also initiated at the jet pump holddown beams, the first instance occurring in a BWR/3 in 1980. Cracking also was found in the beams of two BWR/6 plants. Subsequently, the beams have been redesigned. Figure 2 shows the general arrangement of a jet pump.

PRA Modification
Three sets of PRA models are available for the Peach Bottom nuclear power plant: the NUREG-1150 models, the Individual Plant Examination (IPE) and Individual Plant Examination for External Events (IPEEE) models, and the Accident Sequence Precursor (ASP) model. The NUREG-1150 PRA models for internal and external events at full power were generated in the late 1980s as part of an NRC-sponsored program to consistently analyze five different nuclear power plants. Generally, the NUREG-1150 studies included limited plant-specific data collection, and the external events analyses were performed with a streamlined and somewhat simplified methodology. In contrast, the IPE (for full-power internal events and internal flooding) was performed in the early 1990s and included more recent plant-specific data and design information. However, the IPEEE, performed in the mid 1990s, utilized screening and seismic margins approaches that did not result in models capable of predicting CDFs from external events. Finally, the ASP model for Peach Bottom is a simplified model compared with the NUREG-1150 and IPE models. The ASP model covers only full-power internal events and is considered to be too simplified to be of much use in this study.

Unfortunately, none of the PRA studies are ideal choices to support the evaluation of IGSCC of BWR internals. The NUREG-1150 studies address the plant design as it existed in the late 1980s and contain very little plant-specific data, and the external events analyses suffer from various deficiencies. In contrast, the IPE/IPEEE studies reflect the plant design as it existed in the early 1990s, but again contain limited plant-specific data. Also, the IPEEE studies did not use methodologies that result in CDF predictions. In general, the ASP model is too simplified for the purposes of this study and does not include external events. Finally, all three types of studies did not address the CDF from low power and shutdown operations.

From past studies of IGSCC of BWR internals, it was expected that initiating events such as recirculation line breaks (RLBs) and seismic events will be important. Therefore, it was important that the Peach Bottom PRA chosen for use in this project be accurate with respect to RLBs and seismic events. Because the IPEEE did not use a methodology that could predict a seismic CDF, the IPE/IPEEE set of models is not appropriate for this project. Therefore, the NUREG-1150 set of PRA models were used. However, in order to update these models, several changes were made.

The existing PRA for the selected plant was modified for the IGSCC-induced failures. The following are examples of the modifications that were made:
1. introducing information from the latest Individual Plant Evaluation (IPE)
2. modifying seismic hazard curves to reflect more up-to-date seismic hazard information
3. including internal and external event PRA branches that were not used in previous PRA studies because of low probabilities
4. separating the main steam line, main feedwater line, and RLBs into three individual events, each with its own probability of occurrence
5. adding an event tree for IGSCC-induced initiating events

These modifications are applicable to all of the potential IGSCC-induced failures of reactor internals. However, the model was first run only with the probabilities of IGSCC-induced failures of jet pump components included.

RESULTS OF JET PUMP STUDY

Calculations that have been performed considered sequences initiated by jet-pump failures. To help establish which sequences would have negligible effects on core-damage frequency (CDF), and to identify the sequences that were more probable in increasing the CDF, parameter studies were conducted with the PRA model to determine which sequences could be screened as negligible contributors to CDF, and with structural analysis to determine probabilities of cascading failures.

The generic BWR CDF for internal events and internal flooding, internal fires, seismic events, and low-power and shutdown operation was estimated to be on the order of 5 x 10^{-5} events/rx/yr. Based on the assumption that a 10% increase in CDF is significant to risk (based on an interpretation of the acceptance guidelines in draft DG-1061),[12] and that there were approximately 50-100 scenarios to evaluate, a screening level for each sequence of 1 x 10^{-7} events/rx/yr was chosen for the preliminary calculations. Sequences involving failure of the reactor protection system and emergency core cooling system were evaluated.

The study included an assessment of IGSCC damage to date that has been detected in the various jet pumps components (Table 1) and the potential targets from a large jet pump loose part that might result in cascading failure sequences (Table 2).

Table 1. IGSCC damage in jet pumps

Jet pump component	IGSCC	Failure	Mitigation	Notes
Inlet riser	Y	N		Detected visually
Riser brace	Possibly	N		Primarily fatigue cracking
Holddown beam	Y	Y	Redesign	Material & size changes
Transition piece	N	N		
Nozzle	N	N		
Inlet mixer, throat	N	N		
Diffuser, adapter	N	N		

Table 2. Potential targets from failed jet pump

Component	IGSCC Damage	Damage Potential from Jet Pump	Safety Concern	Comment on Safety Concern
Baffle plate	N	N	–	
Baffle plate cover	Y	Y	Lower reflood level	Only for recirculation line break
Tie rod	N	Y	RPS, LPCS/HPCS diversion	Could fail core shroud
Core shroud	Y	N	–	Only if tie rods fail
Shroud head bolts	Y	Y	None	Only if upward migration possible; most bolts would have to fail
Core spray line	Y	Y	LPCS failure	Only if upward migration possible
Adjacent jet pump	Y	Y	Lower reflood level	Only for recirculation line break

Structural calculations were performed to assess damage to adjacent components that could result from jet pump failures. Thermal-hydraulic studies were conducted by Scientech to calculate the flow rate through the jet pumps, and the flow velocities in the annular region surrounding the jet pumps. IGSCC history, energy required for failed jet pump parts to migrate to nearby components, and energy required to damage adjacent components were considered.

The results showed that there was insufficient energy for loose jet pump parts to migrate upward to damage components such as the core spray system. Loose jet pump parts could contact adjacent jet pumps, the reactor vessel wall, the baffle plate and covers, the core shroud, and core shroud tie rods. Damage to these components would not be expected except for the following:

1. If the adjacent jet pump, baffle plate cover, and core shroud were already very severely damaged by IGSCC, these components could fail, possibly resulting in further cascading failures
2. The plant chosen for evaluation did not contain core shroud tie rods, but calculations showed that impacts from failed jet pumps should not fail properly installed tie rods

Preliminary results show that the cascading failure of an adjacent jet pump or baffle plate covers would not result in a significant increase in CDF. If a core shroud were very severely damaged, it is expected that inservice inspection would have detected the degradation, and repair methods such as tie rods would have been installed. If a tie rod were impacted by loose jet pump part, the rod would not fail if properly installed.

Loose parts from a failed jet pump might migrate through the lower core plenum, and back up into the core, which could block control rod insertion or block coolant from sufficiently reaching a fuel channel. Although probability estimate of loose parts is very difficult to quantify, the studies show that there is a very low probability of this sequence affecting the CDF.

APPLICATION TO OTHER COMPONENTS

Once the methodology was developed, it was applied to the remaining components. The 148 sequences identified in the initial phase of the project included any plausible damage from any of the reactor internals for all BWR types, and without regard to safety analysis or PRA results. In order to reduce the number of sequences from 148 to a more manageable number, the following results and assumptions were used to reduce the list to about 36:

1. A number of the sequences did not affect BWR/4s. This reduced the list by 9.
2. A number could immediately be screened from the plant-specific PRA. This reduced the list by 18.
3. A number of the sequences involved loose parts from various postulated failed reactor internals breaking up, migrating to the core, and entering the small holes in the CRD housings to inhibit control rod insertion or block fuel channels. Studies showed that while probabilities for these sequences were very difficult to quantify, the overall assessment was that damage to a sufficient number of control rod drive mechanisms or fuel elements was below the screening value. This reduced the list by about 45.
4. Safety studies have shown that for core damage to occur, approximately 1/3 of the control rods would have to fail to insert on a random basis, or 5 to 10 control rods grouped together.[13,14] Since a number of the sequences involved only a single rod failure, this reduced the list by about 20.
5. For some events that involved a SBLOCA, the plant makeup system could easily supply makeup during the subsequent plant shutdown, and the event could be screened. This reduced by list by 6.
6. A number of the remaining sequences had been evaluated during the jet pump study. This reduced the list by an additional 17.

Of the remaining sequences, only a few are considered major, and involve either a reactivity accident caused by failure of rods to insert because of misalignment of core internals, or diversion or loss of coolant to the core. These involve the most important internals, such as the core shroud, core plate, top guide, and core spray system. The accident initiators are line breaks and seismic events.

The analyses are dependent on the frequency and magnitude of seismic events and the frequency of line breaks. The mean frequencies for various ground level accelerations were taken from the Peach Bottom seismic hazard curve. The accelerations at the levels of the reactor internals supports were calculated by multiplying the ground level accelerations factors which account for the amplification between ground level and the elevation at the supports, based on a simplified approach recommended by Dr. R.P. Kennedy, a consultant to the project. Various estimates of pipe break probabilities have been made over the years. The values in NUREG-1150[15,16] have received widespread review and have been used in many studies. A recent study by the INEEL[17] recommended lower probabilities (events/rx-yr) for line breaks in U.S. BWR plants. BWRVIP studies[18] recommend an even lower probability for a BWR recirculation system LBLOCA. Sensitivity studies are being included to assess the results from using the various estimates.

Table 3. Estimates of BWR recirculation line break frequency (events/yr)

Source	Mean frequency
NUREG-1150	1E-4
Poloski et al.	2E-5
BWRVIP	7.5E-6

The probability of component failure can be simply stated by the following:

Probability of Failure = Pci x Pmbi x Pgtf

Where Pci = probability of crack initiation
Pmbi = probability crack missed by inspection
Pgtf = probability of crack growth through wall until fragility depth is reached for loading condition under study

A conditional failure probability of 1 could be used, but this would result in a significant increase in CDF for the failure of several of the major components and is too conservative. However, this serves to emphasize that if the degradation if left unmanaged, there could be a significant increase in the probability of CDF.

Inspection crack detection probabilities, crack growth rates, and structural calculations are currently being used to provide estimates of these probabilities as input to the PRA. Figure 3 shows the structural finite element model being used in the calculations. The cutaway view shows the reactor vessel and support, the core shroud and head, the top guide, and the core plate. For clarity, the detailed finite element mesh is not shown.

SUMMARY

A program is underway to perform an independent risk assessment of accident sequences initiated by IGSCC-induced failures of BWR reactor internals components. An initial phase has been completed in which background material was gathered and evaluated, potential accident sequences were identified, and a qualitative PRA was performed to rank the sequences as having a high, medium, or low potential to significantly change the core damage frequency. A second phase is underway to perform a simplified, quantitative PRA on a representative high-power BWR/4. The existing PRA for the plant has been upgraded and modified for the project, including introducing an event tree associated with reactor internals failures. Failures associated with jet pumps were first to establish an analysis methodology. Preliminary results show low probabilities for sequences initiating from jet pump failures significantly affecting the CDF. The methodology is being extended to other major reactor internals components.

ACKNOWLEDGEMENTS
Joel Page was NRC program manager for the initial stage of the program. Steve Eide and Mike Calley of the INEEL performed the PRA analyses, and Mike Nitzel and Tom Clark performed the structural analysis. The staff of the Peach Bottom plant supplied drawings and information to assist in carrying out the project. The Scientech staff conducted thermal-hydraulic calculations for the studies. Dr. R.P. Kennedy served as a structural risk assessment consultant to the project, and Drs. R.W. Staehle and G.W. Was provided consultation on crack growth rate estimates.

REFERENCES

1. United States Nuclear Regulatory Commission, Information Notice 93-79, "Core Shroud Cracking at Beltline Region Welds in Boiling Water Reactors", September 30, 1993.
2. United States Nuclear Regulatory Commission, Generic Letter 94-03, "Intergranular Stress Corrosion Cracking in Boiling Water Reactors", June 25, 1994.
3. United States Nuclear Regulatory Commission, Information Notice 94-42, "Cracking in the Lower Region of the Core Shroud in Boiling Water Reactors", June 7; Supplement 1, July 19, 1994.
4. Medoff, J., *Status Report: Intergranular Stress Corrosion Cracking of BWR Core Shrouds and Other Internal Components*, NUREG-1544, 1996.
5. United States Nuclear Regulatory Commission, Information Notice 80-07, "BWR Jet Pump Assembly Failure", April 4, 1980.
6. United States Nuclear Regulatory Commission, IE Bulletin 80-13, "Cracking in Core Spray Spargers", May 12, 1980.
7. United States Nuclear Regulatory Commission, Information Notice 88-03, "Cracks in Shroud Support Access Hole Cover Welds", February 2, 1988.
8. United States Nuclear Regulatory Commission, Information Notice 92-57, "Radial Cracking of Shroud Support Access Hole Cover Welds", August 11, 1992.
9. United States Nuclear Regulatory Commission, Information Notice 93-101, "Jet Pump Hold-Down Beam Failure", December 17, 1993.
10. United States Nuclear Regulatory Commission, Information Notice 95-17, "Reactor Vessel Top Guide and Core Plate Cracking", March 10, 1995.
11. United States Nuclear Regulatory Commission, Information Notice 97-02, "Cracks Found in Jet Pump Riser Assembly at Boiling Water Reactors", February 6, 1997.
12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis", July 1998.
13. United States Nuclear Regulatory Commission, *Anticipated Transients Without Scram for Light Water Reactors,* NUREG-0460, Vol. 2, 1978.
14. General Electric Company, *Technical Specification Improvement Analyses for BWR Reactor Protection System,* NEDC-30851P, 1985.
15. United States Nuclear Regulatory Commission, *Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants (Final Summary Report),* NUREG-1150, Vol. 1 and 2, 1990.
16. Kolaczkowski, A.M., et al., *Analysis of Core Damage Frequency: Peach Bottom, Unit 2 Internal Events,* NUREG/CR-4550, SAND86-2084, Vol, 4, Rev., 1, Part 1, 1989.
17. Poloski, J.P., et al., *Rates of Initiating Events at U.S. Nuclear Power Plants: 1987-1995,* NUREG/CR-5750, INEEL/EXT-98-00401, 1999.
18. BWRVIP, V. Wagoner to C.E. Carpenter (NRC) letter "BWRVIP Response to NRC Request for Additional Information on BWRVIP-28", dated Dec. 8, 1997.

Figure 1. BWR/3-BWR/4 general arrangement

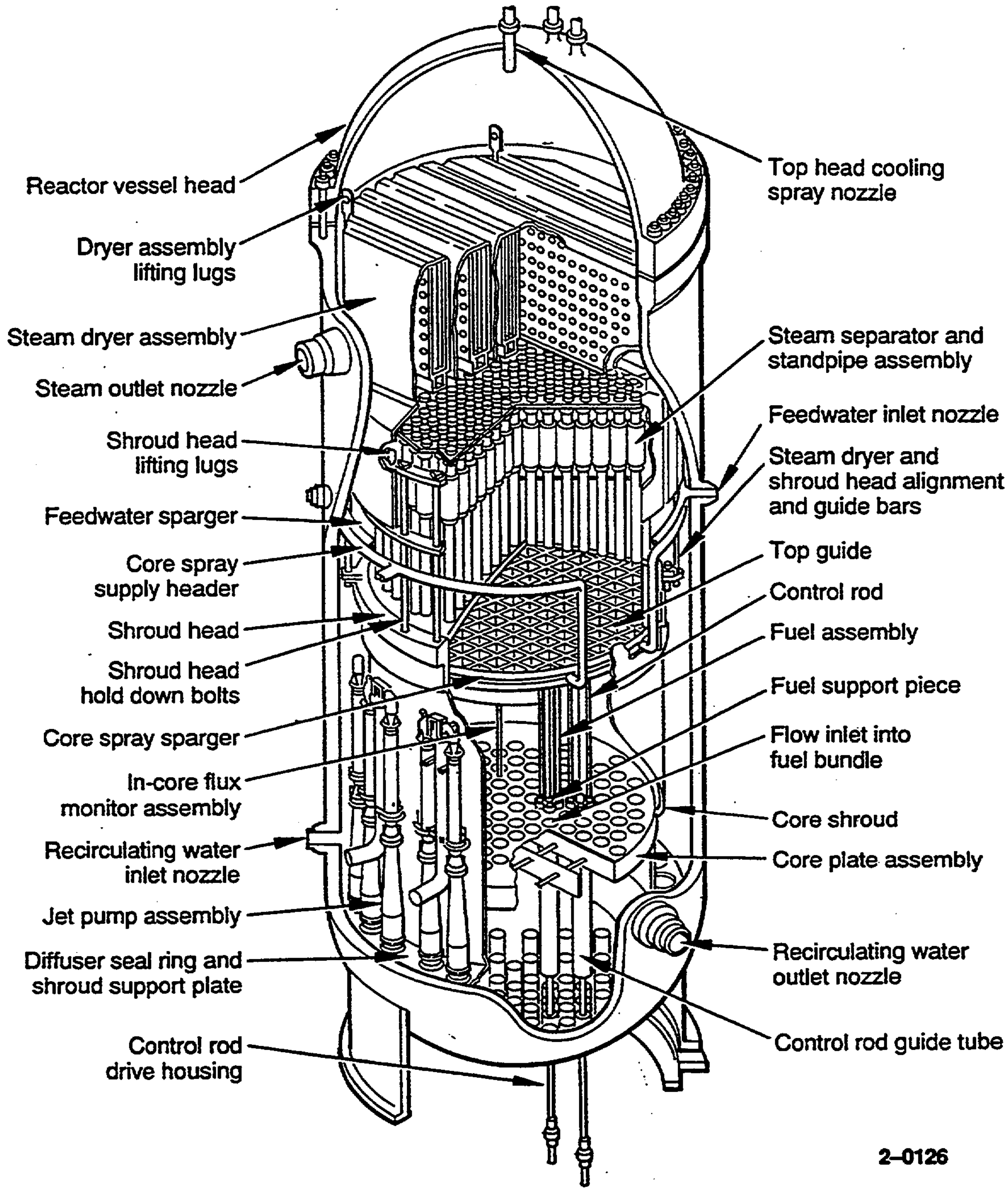


Figure 2. Typical jet pump arrangement

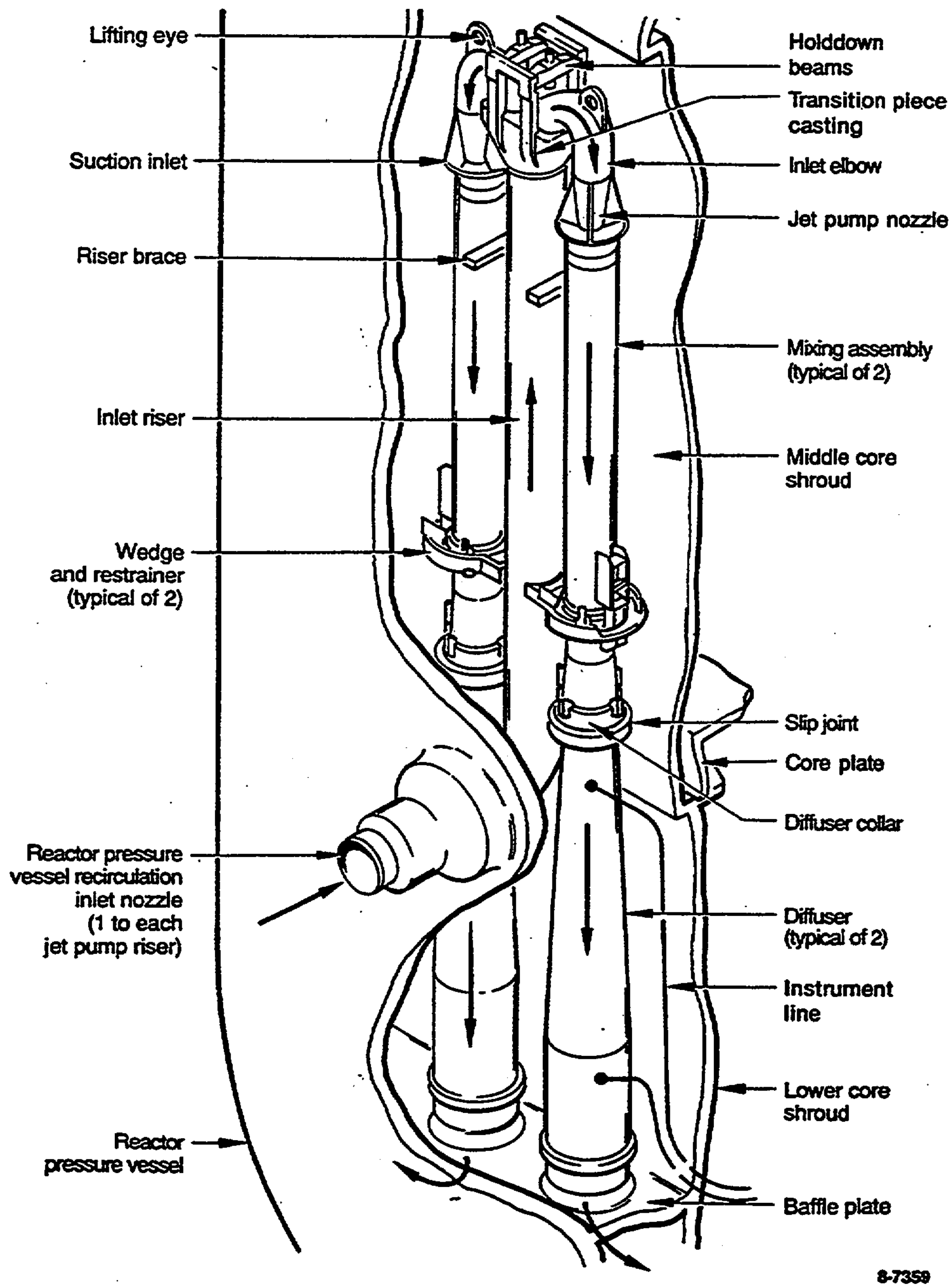
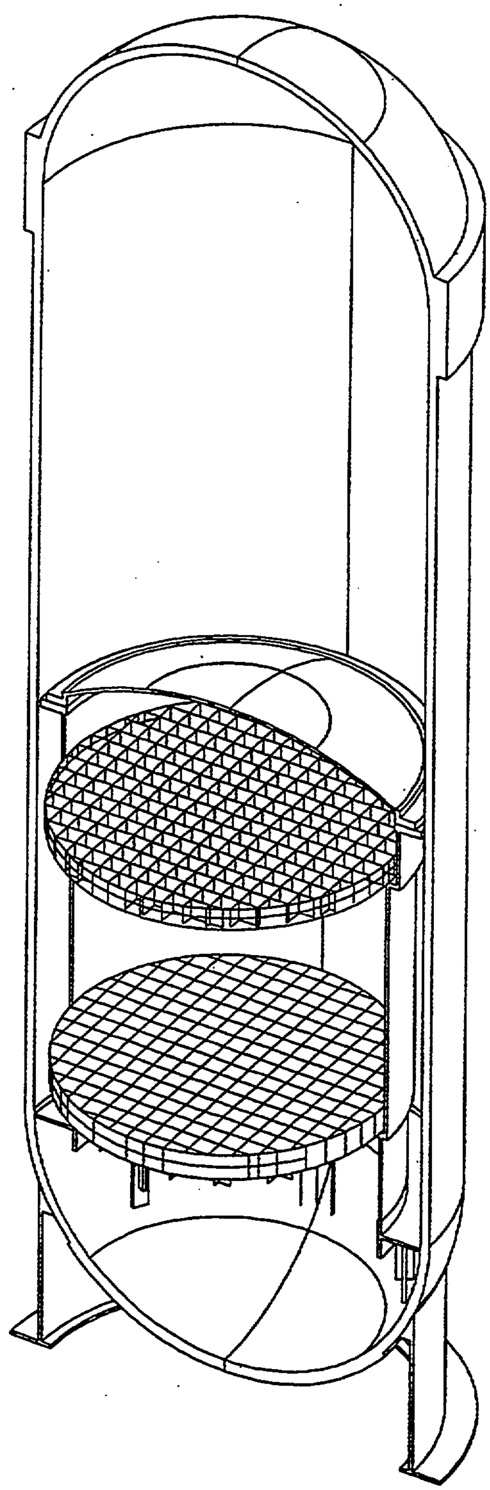


Figure 3. General cutaway view of finite element model, showing reactor vessel and support, core shroud and head, top guide, and core plate



Evaluation of Environmental Effects on Fatigue Life of Piping

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Abstract

Recent data indicate that the effects of light water reactor environments can significantly reduce the fatigue resistance of materials, and show that design fatigue curves may not be conservative for reactor coolant environments. Using revised fatigue curves developed by Argonne National Laboratory (ANL), the work of this paper calculates the expected probabilities of fatigue failures and associated core damage frequencies at a 40-year and 60-year plant life for a sample of components from five PWR and two BWR plants. These calculations were made possible by the development of an enhanced version of the pc-PRAISE probabilistic fracture mechanics code that has the ability to simulate the initiation of fatigue cracks followed by the linking of these cracks. Results of interim calculations subject to review are presented. Components with the highest probabilities of failure can have predicted frequencies of through-wall cracks on the order of about $5x10^{-2}$ per year. The corresponding maximum contributions to core damage frequencies are on the order of 10^{-6} per year. Components with the very high failure rates show essentially no increase in calculated core damage frequency from 40 to 60 years.

Introduction

The American Society of Mechanical Engineers (ASME) Code Section III requires a fatigue evaluation of the components of the reactor coolant pressure boundary. Recent test data indicate that the effects of light water reactor (LWR) environments could significantly reduce the fatigue resistance of materials, and show that the ASME design fatigue curves may not be conservative for nuclear power plant primary system environments. To address this concern, Argonne National Laboratory (ANL) has developed revised fatigue curves based on laboratory test data from small, polished specimens cycled to failure in water with temperatures, pressures and chemistries that simulated LWR conditions (NUREG/CR-6335). ANL has also developed statistical models for estimating the effects of various material, loading, and environmental conditions on the fatigue life of these materials. Fatigue S-N data for carbon steel (CS) and low-alloy steel (LAS), and austenitic stainless steels (SS) were published in NUREG/CR-6335.

Using curves developed by ANL, the Idaho National Engineering Laboratory (INEL) investigated the significance of the interim fatigue curves as published in NUREG/CR-5999 by performing deterministic fatigue evaluations for a sample of components in the reactor coolant pressure boundary of LWRs. Cumulative usage factors (CUF) were reported by INEL in NUREG/CR-6260. The objective of the present work was to calculate component failure probabilities rather than fatigue usage factors. The first such evaluations of risk of failure from fatigue of various reactor coolant system components were performed by NRC staff under Generic Safety Issue GSI-78, ("Monitoring of Fatigue Transient Limits for the Reactor Coolant System"). However, these fatigue analyses assumed a 40-year plant life and used the fatigue life data from NUREG/CR-6237. The objective of the present work was to perform calculations necessary to determine probabilities of fatigue failure for the selected LWR components and to address a 60-year plant life (versus 40-year life) using the most recent fatigue data reported in NUREG/CR-6335 and updated by ANL (Chopra 1998). The present study was based on stress inputs that were extracted from the information presented in NUREG/CR-6260, which in turn were based on conservatively calculated stress values given in design stress reports for the plants.

The ANL data provided the needed statistical model of the number of cycles to crack initiation. All calculations in the present report have assumed that the ANL correlations for crack initiation correspond to a 3-mm crack in a fatigue specimen consistent with a load drop method to detect the presence of cracking. It was then assumed that small initiated fatigue cracks grow based on fracture mechanics rules. The probabilities of crack initiation and that a 3-mm crack becomes a through-wall crack were computed using an enhanced version of the Probabilistic Fracture Mechanics Code for Piping Reliability Analysis (pc-PRAISE code) as described in NUREG/CR-5864. In the improved model the fatigue cracks can initiate at multiple sites around the circumference of a pipe, and can subsequently link to potentially form cracking around the full circumference of the pipe.

A final part of the study estimated the consequences (i.e., core damage frequencies) of the through-wall cracks. Conditional probabilities that a through-wall crack results in small or large leak rates were first estimated. This was followed by an evaluation based on published PRA data regarding core damage for small and large loss-of-coolant accidents (LOCAs). These risk evaluations were performed in a conservative and bounding manner. The objective was to demonstrate that the components of concern are expected to make insignificant contributions to core damage.

This paper begins with a description of the probabilistic fracture mechanics methodology and then describes the plants and components that are addressed by the fatigue analyses. Results are then given for probabilities of through-wall cracks and for small and large leaks. The discussion addresses the consequences of small and large leaks and describes how calculations were performed to estimate core damage frequencies. Results are summarized in terms of absolute and relative failure probabilities, giving particular attention to how these calculated probabilities differ for air versus water environment and for a 40-year versus 60-year plant life. All information presented in this paper from component life predictions and from estimates of core damage frequencies should be considered interim results subject to further review.

Probabilistic Fracture Mechanics Methodology

Probabilistic fracture mechanics calculations were performed to estimate the probability that a given combination of cyclic fatigue stresses will result in through-wall cracks in pressure boundary components of reactor coolant systems. These evaluations addressed only the contribution of initiated fatigue cracks, and exclude the contributions of preexisting cracks. The methodology first calculated the probability that a fatigue crack will initiate, and then evaluated the probability that such cracks would grow to become through-wall.

In the probabilistic fracture mechanics calculations, the crack propagation was assumed to start from a 3-mm deep initiated flaw. This size was based on the estimated crack size that can give a measurable load drop in the testing of standard fatigue specimens. Sensitivity calculations were performed to evaluate the effect of changing this crack depth to 2-mm or 4-mm. The resulting changes in calculated probabilities of through-wall cracks were about a factor of two or less.

Calculations were performed with an enhanced version of the pc-PRAISE code (Harris and Dedhia 1992) that expanded the capabilities of the code to simulate fatigue crack initiation in addition to a simulation of fatigue failures due to preexisting fabrication flaws. The enhanced code addresses crack initiation at multiple sites by subdividing the pipe circumference into a set of 2-inch long zones. The amplitude of cyclic stresses at each site can vary in a systematic manner as specified by user input such that the fatigue cracks can potentially initiate at some sites much sooner than at other sites. The model assumes no correlation between the random scatter in crack initiation times from one site to the next. Thus a different selection from the population of SN curves is sampled at random for each of the various sites. There is, however, an option (not used for the present calculations) to assume that the SN curves for all sites are perfectly correlated. This option would very conservatively predict that a fatigue crack initiates at all of the circumferential sites at precisely the same number of stress cycles.

The probabilistic fracture mechanics capabilities permitted a number of important issues to be addressed. It is not necessary to conservatively assume that cracks have the full service life of the plant to grow to through-wall depths. Rather the simulations start the growth of the cracks at whatever time the cracks are predicted to initiate. The effects of through-wall stress gradients on the growth of the initiated cracks are included in the calculations. The starting lengths of the initiated fatigue cracks are addressed along with a simulation of the subsequent cycle-by-cycle changes in the crack lengths during the crack growth process. The model also simulates the initiation of fatigue cracks at multiple sites around the pipe circumference, and predicts the formation of longer cracks by the linking of cracks in adjacent initiation sites.

The cyclic stress levels from the INEL report (NUREG/CR-6260) were used to calculate both fatigue usage factors and probabilities of crack initiation. The stresses include the effects of stress concentrations in a manner prescribed by the ASME Code approach of stress indices. In many cases the stress indices may address very high local stresses (e.g., weld-root stress concentrations) and have values up to 2.0. It was recognized that such surface stresses are not indicative of internal stress levels remote from the stress concentration. The present crack growth calculations with pc-PRAISE are based on the same inputs for cyclic stresses as used for crack initiation calculations. Adjustments were made to crack-tip stress intensity factors for deeper cracks to account for through-wall stress gradients that are characteristic of thermal type transients. However, the fracture mechanics calculations may be conservative for many locations, because the local stress levels from stress concentrations would have larger stress gradients than those from thermal transients.

The present crack growth calculations assumed that the random variations in fatigue crack growth rates were uncorrelated with the corresponding variations in the cycles to crack initiation. If such correlations are significant, the predictions for probabilities of through-wall cracks could be somewhat unconservative. The simplifying assumption greatly facilitated the calculations, and has a good technical basis because the technical literature (Wire and Li 1996) supports the assumption of independence. In general, crack initiation and crack growth involve independent material damage mechanisms, such that the factors of environment and loading rates affect the mechanisms for crack initiation and crack growth differently.

Fatigue Crack Initiation Model

The present work used a crack initiation model (NUREG/CR-6335) developed by ANL. This model estimates the probability of initiating a 3-mm deep fatigue crack based on existing fatigue strain versus life (S-N) data, foreign and domestic, for carbon, low-alloy and stainless steels used in the construction of nuclear power plant components. Only data obtained on smooth specimens tested under fully reversed loading conditions were considered. A statistical distribution was fitted by ANL to the strain-life (S-N) data to describe the scatter in the fatigue data.

The ANL statistical distributions of cycles to initiate a 3-mm crack for a given cyclic stress were lognormal. The parameters of the probabilistic fatigue initiation curves were based on the ANL revised fatigue curves published in NUREG/CR-6335. The equations for stainless steels included recent updates for fatigue life correlations provided by ANL (Chopra 1998).

The ANL equations for the number of cycles (N_i) to crack initiation for low alloy steels (LAS) and carbon steels (CS) are given here to illustrate the methodology. The equations for 306, 316 and 316NG follow a similar format. For LAS and CS (both water and air environments) the number of cycles to crack initiation is expressed by (NUREG/CR-6335) as

$$\begin{aligned}\ln[N_i(x)] = \; & (6.857 - 0.766\, I_W) - (0.275 - 0.382\, I_W)\, I_S + 0.52\, F^{-1}[x] \\ & - (1.813 + 0.219\, I_S) \ln(\varepsilon_a - 0.080 - 0.014\, I_S + 0.026\, F^{-1}[1-x]) \\ & -0.00133\, T\,(1 - I_W) + 0.1097\, S^* T^* O^* \varepsilon^* \;\; - \ln(4)\end{aligned}$$

where:

ε_a = the applied strain amplitude, %,

I_w = indicator for water environment. It is 1 for water and 0 for air environment,

I_S = indicator for steel type equal to 1 for carbon steel and 0 for low-alloy steel,

T = the test temperature in °C. The variables S^*, T^*, O^*, ε^* are transformed sulfur content, temperature, dissolved oxygen (DO), and strain rate, respectively, defined as follows:

$$S^{*} = \begin{cases} S & 0 < S < 0.015 \quad wt.\% \\ 0.015 & S > 0.015 \quad wt.\% \end{cases}$$

$$T^{*} = \begin{cases} 0 & T < 150°C \\ T - 150 & T > 150°C \end{cases}$$

$$O^{*} = \begin{cases} 0 & DO < 0.05\,ppm \\ DO & 0.05\,ppm \le DO \le 0.5\,ppm \\ 0.5) & DO > 0.5\,ppm \end{cases}$$

$$\dot{\varepsilon}^{*} = \begin{cases} 0 & \dot{\varepsilon} > 1\%s \\ \ln(\dot{\varepsilon}) & 0.001 \le \dot{\varepsilon} \le 1\%s \\ \ln(0.001) & \dot{\varepsilon} < 0.001\%s \end{cases}$$

The functions $F^{-1}[x]$ and $F^{-1}[1-x]$ are the inverse of the standard normal cumulative distribution function. The constant 0.1097 replaces the value of 0.554 of Equation 18 of NUREG/CR-6335 as per a communication of June 25, 1996 from J.Keisler of ANL to M.A. Khaleel of PNNL.

The term ln(4) was introduced by ANL to apply the fatigue data from small test specimens to full size components. This term applies a reduction factor of 4.0 to the number of cycles to failure to account for size effects, surface finish and geometry. The factor of 4.0 was selected because it gave a relatively good correlation between the small specimen data and published results of fatigue experiments performed on small (9-inch diameter) pressure vessels.

The above equations for cycles to failure were coded into a Fortran subroutine for implementation into the probabilistic fracture mechanic codes such as pc-PRAISE. The calling program needs to provide values for the stress amplitude, the material type, the sulfur content (for ferritic steels), temperature, whether the environment is water or air, oxygen content of the water, and the strain rate for the stress cycle. A final parameter is a percentile value that describes the fatigue life of the simulated component relative to the median fatigue curve. Figure 1 was generated from data obtained from a series of calls to the subroutine. Each of the curves corresponds to the indicated percentile of data having cycles to failure less than or equal to the indicated percentile. The solid curve of Figure 1 is the median (or 50th percentile) curve for cycles to crack initiation.

When implemented into a Monte Carlo simulation, a random number (between zero and one) is sampled before the call to the subroutine to simulate the percentile of the SN curve to be used to predict crack initiation at the particular structural location of concern. This curve is assumed to apply to all the cyclic stress transients for that location with a different curve selected on the basis of the random numbers generated for each of the Monte Carlo simulations.

The enhanced version of pc-PRAISE addresses crack initiation at multiple sites by subdividing the pipe circumference into 2-inch long zones. The amplitudes of cyclic stresses at each site can vary in a manner specified by user input such that the fatigue cracks may initiate at some sites much sooner or later than at other sites. The model also assumes no correlation between the random scatter in crack initiation times from one site to the next. Thus a different selection from the family of SN curves (as shown by the example of Figure 1) is sampled at random for each of the various sites around the pipe circumference. There is also an option (not used for the present calculations) that assumes that the SN curves for all sites are perfectly correlated. This option would predict (rather unrealistically) that a fatigue crack initiates at each of the sites around the pipe circumference after precisely the same number of stress cycles.

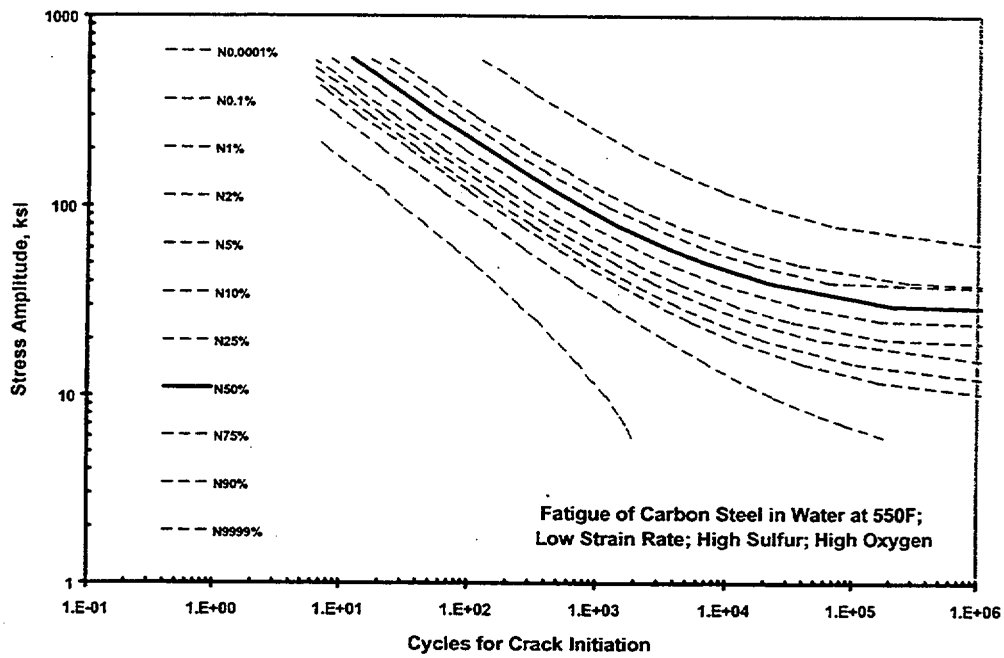

Figure 1 Example of Probabilistic S-N Curves for Low Alloy Steel

Fatigue Crack Growth Model

The fracture mechanics model of pc-PRAISE calculates the probability of a through-wall crack, given that a 3-mm crack has initiated during the plant operating period. The model for fatigue crack growth is the same model used in prior versions of pc-PRAISE to address fatigue failures caused by preexisting fabrication defects. The main difference is that the crack growth begins sometime during the component's life rather than at the beginning of life. The sizes of the initial flaws are consistent with initiated fatigue cracks rather than welding defects.

The growth of a two-dimensional, semi-elliptical circumferential crack at the inner surface of a component is simulated. The initial depth of the crack is always 3-mm, but the length of the defect is sampled from a statistical distribution. Both the lengths and depths of the cracks are allowed to grow. A leak occurs if the crack grows through the entire wall pipe.

The stress at the inner surface of the pipe governing crack growth is the same cyclic stress used for crack initiation. However the model allows for the attenuation of the high surface stresses and thereby accounts for through-wall stress gradients and /or stress concentrations.

In the case of a 360-degree circumferential crack, the stress intensity factor is given by

$$K = \sigma \, (\pi \, a)^{1/2} \, F \left(\frac{a}{h}, \frac{R_i}{R_o} \right)$$

where σ, h, a, R_i, and R_o are the stress, wall thickness, crack depth, internal radius, and external radius, respectively. The function F is obtained by the finite element or other numerical methods. The calculation of the stress intensity factor for surface flaws is based on the contributions of thermal and pressure stresses. For cracks with a finite aspect ratio (i.e., b/a less than 100), the stress intensity factors become lower as the flaw aspect ratios become smaller.

Stress intensity factor solutions generated by the pc-PRAISE code were compared to recent published solutions from the technical literature. This review indicated that a great deal of information on stress intensity factors has become available since the last improvements were made in 1984 to the influence functions of pc-PRAISE. Of particular concern was the behavior of the pc-PRAISE solutions for very long circumferential cracks. It was concluded that the stress intensity factor solutions in pc-PRAISE are well behaved for very long and very deep cracks. The largest uncertainties are associated with stress intensity factors at the surface location for the finite length flaws. Comparisons were somewhat difficult because pc-PRAISE uses root mean square (RMS) values based on energy release rates for stress intensity factors, whereas most of the literature uses local values. The surface values are important because they control the lengthwise growth of the cracks, which has a large influence on the final crack lengths that govern calculated leak rates at the time that the cracks penetrate the outer surface of the pipe.

Another review addressed the proposed changes being made to the methodology of the ASME Section XI code for predicting the changes in the shapes of growing fatigue cracks. These code changes are based in large part on experimental results provide by Professor Iida from Japan. The Japanese experiments show that the final shape of a fatigue crack (i.e., when the crack penetrates the pipe wall) has an aspect ratio (ratio of total crack length to the crack depth) in the range of 2 to 4. These experimental values are consistent with aspect ratios predicted by pc-PRAISE. Such agreement provides indirect support to the stress intensity factor solutions in the code. The experimental trends do not preclude the development of very long fatigue cracks, because long fatigue cracks can also result from the linking of several individual cracks.

Fatigue crack growth can be described by the modified Forman relation (Forman et al. 1988), which is a general functional form for curve-fitting fatigue crack growth data. In addition, the well-known Paris relationship has been found by many researchers to provide a good fit for a wide variety of materials. Article A-4000 of the ASME Section XI Code relates the fatigue crack growth rate da/dN of a material to the range of applied stress intensity factor ΔK. A probabilistic form of the ASME equations was used in the present probabilistic fracture mechanics (PFM) model for the crack growth rates. For low alloy steels subject to water environments the fatigue crack growth rate da/dN (inch per cycle) is:

$$\frac{da}{dN} = Z \begin{cases} 1.03 \; x \; 10^{-12} \; S \; (\Delta \; K)^{5.95} & \Delta K \leq K_{knee} \\ 1.01 \; x \; 10^{-7} \; S \; (\Delta \; K)^{1.95} & \Delta K > K_{knee} \end{cases}$$

The term K_{knee} , in the above equation is

$$K_{knee} = \begin{cases} 17.74 & R \le 0.25 \\ 17.74\left(\dfrac{3.75R + 0.06}{26.9R - 5.725}\right)^{0.25} & 0.25 < R < 0.65 \\ 12.04 & R \ge 0.65 \end{cases}$$

If $K < K_{knee}$ the adjustment factor S is

$$S = \begin{cases} 1.0 & R \le 0.25 \\ 26.9\,R - 5.725 & 0.25 < R < 0.65 \\ 11.76 & R \ge 0.65 \end{cases}$$

while if $K > K_{knee}$, the factor is

$$S = \begin{cases} 1.0 & R \le 0.25 \\ 3.75\,R + 0.06 & 0.25 \le R < 0.65 \\ 2.5 & R \ge 0.65 \end{cases}$$

The parameter R, which accounts for mean stress effects on crack growth rates, is defined in terms of the minimum and maximum stress intensity factors during the stress cycle as $R = K_{min}/K_{max}$. The parameter Z is added to the crack growth equation to randomize the crack growth rates. This random variable covers all possible uncertain quantities such as material variability, environmental variability, crack geometry variability, and crack modeling uncertainty. The parameter Z is assumed to have a lognormal distribution.

The fatigue crack growth rate (da/dN in inches per cycle) for austenitic stainless steel piping is represented by the following relation:

$$\frac{da}{dN} = C\left[\frac{K_{max} - K_{min}}{\left(1 - \dfrac{K_{min}}{K_{max}}\right)^{1/2}}\right]$$

where K_{min} and K_{max} are the minimum and maximum stress intensity factors (ksi-in$^{1/2}$), respectively. The scatter in the data is represented by a lognormal value of C with a median of 9.14×10^{-12} and standard deviation of 2.20×10^{-11}.

Example Calculations for Surge Line Elbow

Example calculations for a component with high fatigue usage were performed for the surge line elbow of the newer vintage Combustion Engineering plant. The results for this component are compared with results for another component selected from the same plant that has much lower levels of fatigue usage.

Figure 2 presents the failure probabilities predicted by pc-PRAISE for the surge line elbow in terms of probabilities of crack initiation and through-wall cracks as a function of time. It is seen that cracks initiate rather early in the plant life. There is about a 50 percent probability for a fatigue crack after only ten years of operation. Over this 10-year time interval, about 50 percent of the initiated cracks are predicted to grow to become leaking cracks. The frequency of through-wall cracks (lower curve) increases significantly over this 10-year period, but then remains relatively constant over the remainder of the 60-year plant life. These results would indicate a relatively constant failure rate for locations that have relatively high levels of fatigue usage.

Figure 3 addresses another component (reactor pressure vessel outlet nozzle for the newer vintage Combustion Engineering plant) that has a much lower level of fatigue usage. The failure rates for this component continue to increase significantly over the entire 60-year plant life. However, the maximum rates never approach the very high rates predicted for the surge line elbow.

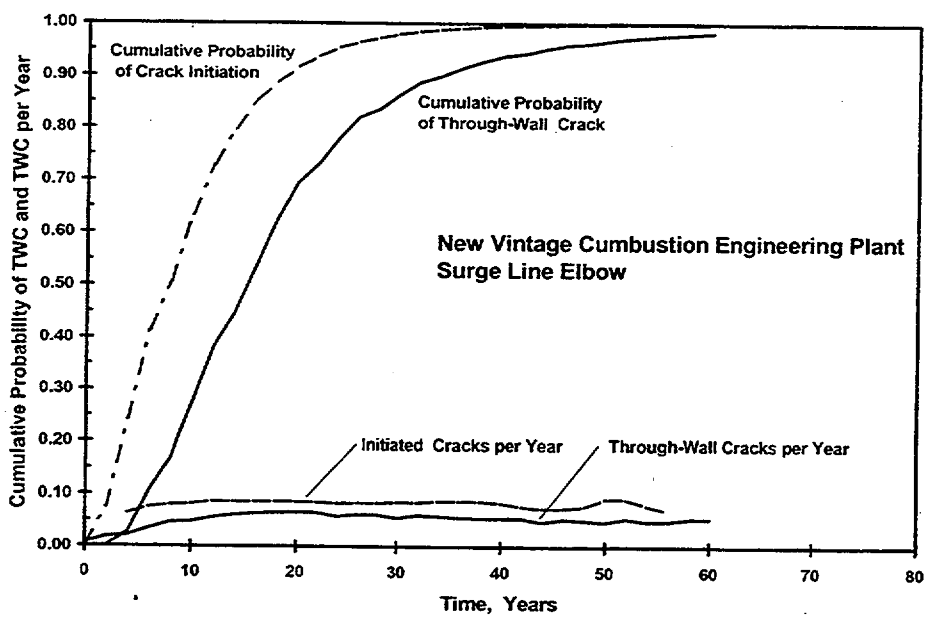

Figure 2 Calculated Probabilities of Crack Initiation and Through-Wall Crack for the Surge Line Elbow of the Newer Vintage Combustion Engineering Plant

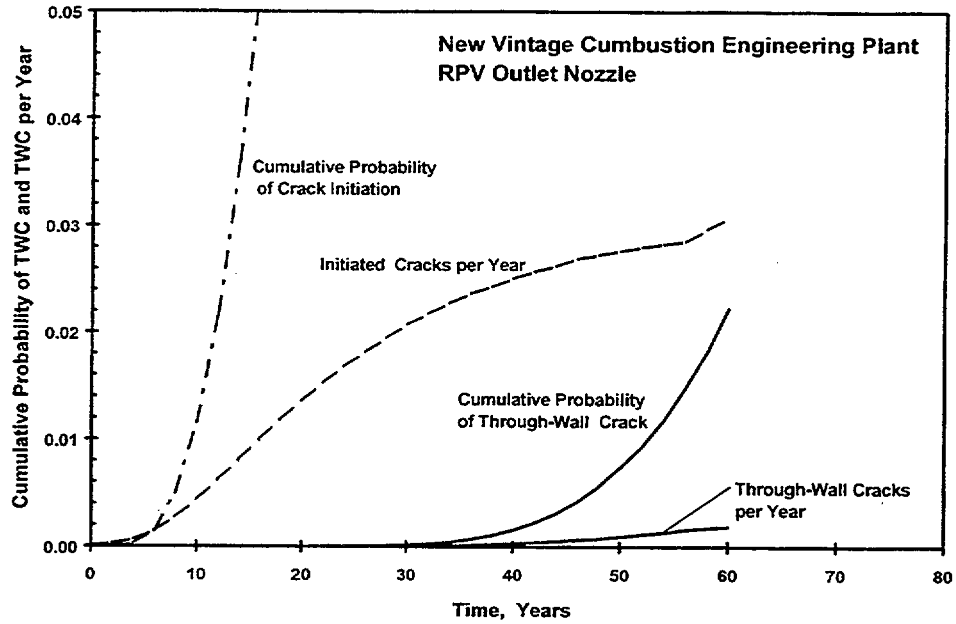

Figure 3 Calculated Probabilities of Crack Initiation and Through-Wall Crack for the RPV Outlet Nozzle of the Newer Vintage Combustion Engineering Plant

Conditional Probabilities of Small and Large Leak Rates

The fracture mechanics calculations predicted probabilities that initiated fatigue cracks will become through-wall flaws. In most cases these flaws will initially be relatively short and will result in only minor leakage such as 3 gallons per minute (gpm) with no safety consequences. The leakage will tend to increase over time as the crack continues to grow, which means that the leakage will eventually reach detectable levels that results in plant shutdown before the leak rates become sufficiently large to potentially impact plant safety. Nevertheless, some (small) fraction of the through-wall cracks can be relatively long from the onset, and could leak immediately at larger rates that are sufficient to be of concern to plant safety. While leak detection measures would not themselves mitigate the effects of such cracks, injection systems would compensate for the losses of reactor coolant. Safety consequences would occur only if these normally reliable systems fail to function as intended.

An important step in the evaluation was to estimate the probability that a given through-wall crack would leak at a rate sufficient to cause a loss-of-coolant accident. The objective was to estimate conditional probabilities that the leakage from a through-wall crack would be at the rates corresponding to these predefined categories.

The leakage categories were selected as

1) less than 30 gpm,
2) 30 gpm to 500 gpm,
3) greater than 500 gpm.

These categories corresponded to the categories used in the Westinghouse Owners Group/Virginia Power pilot application of risk-informed inspection (WCAP 14572 Revision 1).

Although the pc-PRAISE fracture mechanics model can predict probabilities for prescribed leakage rates, the present study did not perform such component specific calculations. The available information did not include sufficient data on loads and stresses to support such calculations. Furthermore, many of the component geometries (e.g., nozzle configurations) did not correspond to the pc-PRAISE fracture mechanics model that described only circumferential cracks in piping. The conditional probabilities were therefore estimated by application of trends from sensitivity calculations performed with pc-PRAISE and by reference to service experience with piping failures (frequencies of small versus large leaks).

Data from operating experience at nuclear power plants (Bush et al. 1996) show that the reported numbers of small leaks are many times greater than the numbers for large leaks (or ruptures). Even for mechanisms such as vibrational fatigue and flow-assisted corrosion, the ratio can be as high as 10:1. For other mechanisms such as stress corrosion cracking, the data indicate ratios of small leaks to large leaks of 1000:1 or greater.

There have been thermal fatigue failures that have resulted in relatively long cracks, which can be described as "near misses" for pipe rupture events. The cyclic stresses addressed in the present report can be largely described as thermal fatigue type stresses. These cases show the potential for long cracks that can cause large leaks. Nevertheless, experience shows that even these long cracks tend to have variations in their depths along the crack front, such that one part of the crack front will break through the wall and cause a detectable leak before a pipe break occurs.

The pc-PRAISE code was applied to gain insight into the (small) fraction of through-wall cracks that will cause significant leaks. Such calculations are sensitive to inputs regarding the initial lengths of the flaws and to the assumptions made to predict the lengthwise versus depth-wise growth. The pc-PRAISE model assigned a distribution to the length of the 3-mm deep fatigue cracks that had a median aspect ratio of about 5:1 (ratio of total flaw length to flaw depth). The lognormal distribution of flaw lengths had a probability of 10^{-2} that the length of the initiated crack will extend the full length (2-inch) of the standard initiation site, which corresponds to an initial aspect ratio of about 17:1. The model also allowed for crack initiation at the multiple sites around the circumference of a pipe, and then simulated the possible linking of small cracks in adjacent zones to create much longer cracks. Thus the simulation was capable of predicting the occurrence of the very long cracks such as sometimes observed in service degraded piping.

Figure 4 shows example results from a systematic set of calculations for probabilities of small and large leaks. Results are shown for a 12-inch diameter stainless steel pipe at PWR conditions of 550F and 2,250 psi. Although the probabilities of through-wall cracks approach 100 percent, the corresponding probabilities of large leaks (30, 100 and 500 gpm) are much smaller.

The probabilities for leak rates of 30, 100 and 500 gpm were expressed as conditional failure probabilities. Figure 5 summarizes results for several pipe sizes and leak rates. The conditional probabilities are predicted to decrease as the pipe diameter becomes larger. The vertical arrays of points on Figure 5 correspond probabilities at different times during the 60-year plant life, with the upper points corresponding to the higher conditional probabilities that occur later in the 60-year operating period.

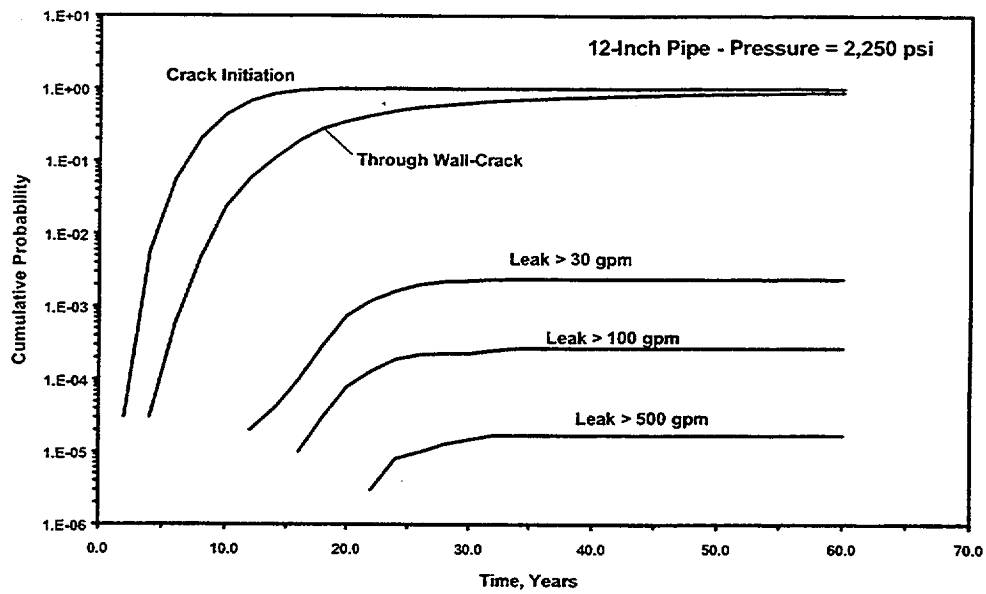

Figure 4 Leak Probabilities Versus Time for 12-inch Diameter Pipe

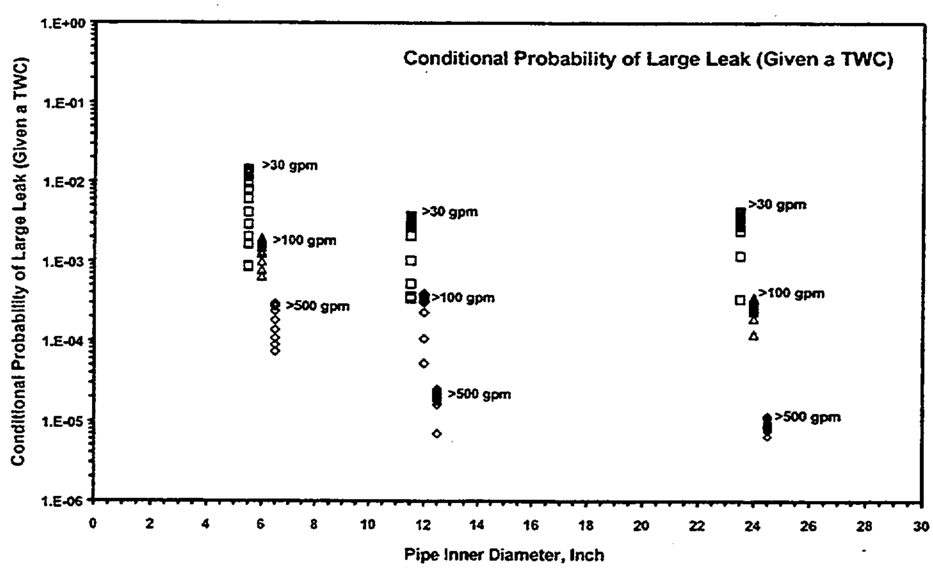

Figure 5 Conditional Probabilities of Small and Large Leaks

Table 1 gives conditional probabilities of small and large leaks used in the subsequent estimates of core damage frequencies. This table was derived from Figure 5 by developing bounding curves for each of the three leak rates (30, 100 and 500 gpm) that corresponded to the upper end of the scatter band of the vertical array of points corresponding to each leak rate. These curves were then adjusted upwards by a factor of 10 to allow for uncertainties in the pc-PRAISE calculations. This adjustment was consistent with the intent to evaluate core damage frequencies in a bounding manner. Based on other sensitivity calculations, the adjustment factor was not applied for nozzle type locations. These nozzle locations should have relatively short through-wall cracks because the high stress gradients act in both the axial and radial directions.

Table 1 Conditional Probabilities of Failure Modes Given the Occurrence of a Through-Wall Crack

	Conditional Probability Given a Through-Wall Crack		
Pipe Diameter (inch)	Through-Wall Crack Leak > 30 gal/min	Through-Wall Crack Leak > 100 gal/min	Through-Wall Crack Leak > 500 gal/min
2	2E-01	8E-02	8E-02
3	1.5E-01	5E-02	4E-03
4	1E-01	4E-02	3E-03
6	8E-02	2E-02	1E-03
8	6E-02	1E-02	6E-04
10	5E-02	7E-03	3.5E-04
> 12	5E-02	5E-03	2E-04

Conditional Core Damage Probabilities

Estimates of conditional core damage probabilities were based on numerical parameters obtained from probabilistic risk assessments (PRAs) for plants that covered all seven plant types addressed by the present fatigue evaluations. Because the evaluations of core damage frequencies were a secondary objective of the study, these calculations were performed in a conservative or bounding manner to establish whether or not the calculated frequencies of through-wall cracks implied significant or negligible contributions to overall plant risk. It was not the intent to make accurate comparisons of core damage frequencies from plant-to-plant or from location-to-location within a given plant. The available PRAs covered the plant vendors and vintages of interest, but there was not a one-to-one match of the plants that were used for the fatigue analyses. Therefore the evaluations were performed on a generic basis using common consequences from location-to-location within each plant and differentiating from plant-to-plant only in terms of PWR and BWR plants.

It was assumed that all fatigue locations were part of the primary coolant system boundary, and that failures could result in a loss of coolant accident (LOCA). Conditional core damage probabilities (given a small or large leak) were established by application of probabilistic risk-assessment (PRA) models. The recently completed study of risk-informed inspection for the Surry-1 plant (WCAP-14572 Revision

1) served as a benchmark for other PRA results. The other PRA evaluations addressed various plants (e.g., the NUREG 1150 sample of plants) and included the Surry-1 plant. The NUREG-1150 results for conditional core damage probabilities for Surry-1 were found to be in good agreement with the more recent WCAP results.

Conditional core damage probabilities were developed for a total of seven plant types, including old and new vintage plants supplied by Combustion Engineering (CE), Westinghouse (W), General Electric (GE), as well as a typical Babcock and Wilcox (B&W) plant. Table 2 lists the plants for which PRA information was applied in the present study. The new and old vintage Combustion Engineering plants were represented by the Palo Verde and Calvert Cliffs PRAs respectively. The Oconee plant was used as a representative B&W plant. New and old vintage Westinghouse plants were covered by the Sequoyah and Surry-1 plants. BWR plants of General Electric designs were addressed by the Grand Gulf (newer vintage) and Peach Bottom (older vintage). Table 2 also lists the break sizes (expressed as equivalent circular hole diameter) that were used in each PRA to define the LOCAs of the small, medium and large categories. In each case the leakage rates (for the pressures and temperatures of normal plant operation) corresponding to each break diameter are also listed.

Table 2 Summary of PRAs in Terms of Plant Vendors, Plant Vintages, Break Sizes, and Leak Rates

	Plant Type						
LOCA Category	New-CE (Palo Verde)	Old-CE (Calvert Cliffs)	B&W (Oconne)	New-W (Sequoyah)	Old-W (Surry)	New-GE (Grand Gulf)	Old-GE (Peach Bottom)
Small LOCA Diameter, Inch	0.38 - 3.0	0.3 -1.9	< 4.0	< 2.0	0.5 – 2.0	< 1.1	< 0.9
Small LOCA gpm	12 -700	8 - 250	< 1200	< 300	20 - 300	< 30	< 25
Medium LOCA Diameter, Inch	3.0 – 6.0	1.9 – 4.3	N/A	2.0 – 6.0	2.0 – 6.0	1.1 – 4.9	0.9 – 4.3
Medium LOCA gpm	700 - 2800	250 - 1400	N/A	300 - 2800	300 - 2800	30 – 750	25 – 550
Large LOCA Diameter, Inch	> 6.0	> 4.3	> 4.0	> 6.0	> 6.0	> 4.9	> 4.3
Large LOCA gpm	> 2800	> 1400	> 1200	> 2800	> 2800	> 750	> 550

The accident sequences initiated by failure of the reactor vessel or other RCS components were modeled here as a loss-of-coolant accident. The break was assumed to be of sufficient size that the RCS partially or completely depressurizes such that the plant system responds to the condition. For example, in the case of larger breaks: 1) the accumulators (or safety injection tanks) inject water immediately into the RCS to make up for the water leaking from the break; 2) low pressure injection (LPI) systems activate to provide additional water for continued core cooling; 3) low pressure recirculation (LPR) cooling from the containment sump is established to provide long-term cooling; and 4) containment pressures and temperatures are maintained by the residual heat removal (RHR) or other safety-related system.

Within the PRAs used for the present calculations, the ruptured RCS components were assumed to disable at least one train of the safety system designed to inject or recirculate coolant. Thus, if there are redundant, 100% capacity trains of LPI, for example, the effect of the break is to reduce the 1-out-of-2 LPI systems to a one-out-of-one system. This is basically how most PRAs model large LOCAs, as most safety injection systems inject into the RCS cold legs, resulting in the inability to inject via the affected nozzle.

In general, the probabilities of unsuccessful response to the initiating events for each size of LOCA were obtained by dividing the overall core damage frequency (CDF) for LOCA sequences by the large LOCA initiator frequency.

Due to the nature of the information available from the PRAs used in the present study, it was not possible to perform detailed modeling to more precisely estimate the conditional probabilities of core damage for specific break locations. The approximations used in the referenced PRAs assumed that the conditional probability of unsuccessful plant response to the initiating event was independent of break location. Furthermore, specific safety system responses were independent of the prior initiating events and successful or unsuccessful operation of all other safety systems. The resulting conditional probabilities are therefore used in the present study as conservative approximations of safety system failure probabilities.

Table 3 summarizes the plant specific PRA results for the seven plant types in terms of conditional core damage probabilities for each category of LOCA. Plant specific definitions of small, medium and large LOCAs are the same as those given in Table 2. In the case of the Surry-1 plant, an additional result for the medium LOCA is given in Table 3. This result shows a core damage probability intermediate to the values for the small and large LOCA categories.

A review of Tables 2 and 3 shows significant differences between results for the PWR plants as a group and the BWR plants as a second group. There are similar trends both in terms of the definitions of break sizes and in the conditional core damage frequencies across the PWR and BWR plant types. It was decided that it was not meaningful to differentiate between specific PWR or BWR plants. Furthermore, the specific plants for the PRA results did not correspond (other than in terms of vendor and vintage) to the (unidentified) plants addressed in the fracture mechanics calculations. Therefore two "generic" categories were used with reference to Tables 2 and 3 to describe the PRA results. The pipe break or leak categories for the PWR and BWR plants were assigned as given in Table 4. The generic categories were selected for convenience to match the leakage rates selected in the probabilistic fracture mechanics calculations. These leakage rates were then mapped in a conservative manner to the LOCA categories of the PRAs. The results of Table 3 were used to assign the generic conditional core damage probabilities of Table 5.

Table 3 Conditional Core Damage Probabilities for PWR and BWR Plants

	Conditional Core Damage Probability		
Plant Type	Small LOCA	Medium LOCA	Large LOCA
New CE (Palo Verde)	4.2E-04	-	4.2E-03
Old CE (Calvert Cliffs)	5.7E-04	-	8.7E-03
B&W (Oconne)	9.2E-05	-	2.7E-03
New W (Sequoyah)	4.8E-04	-	1.9E-02
Old W (Surry-1)	8.4E-04	4.0E-03	1.0E-02
New GE (Grand Gulf)	< 3.3E-08	-	1.1E-05
Old GE (Peach Bottom)	1.6E-06	-	5.3E-04

Table 4 Leak/Break Categories for Generic Treatment of PRA Results for PWR and BWR Plants

Plant Type	Leak With No Safety Consequences	Small LOCA	Large LOCA
PWR	< 30 gpm	30 – 500 gpm	> 500 gpm
BWR	-	> 0 gpm	> 30 gpm

Table 5 Consequences Assigned for Generic Leak/Break Categories for PWR and BWR Plants

Plant Type	Small LOCA	Large LOCA
PWR	5.E-04	1.E-02
BWR	1.E-06	5.E-04

Methodology for Calculating Core Damage Frequencies

Through-wall cracks can cause core damage if the leakage rate exceeds the leakage rates corresponding to plant criteria for small or large loss-of-coolant accidents. The following equations were used to calculate core damage probabilities:

$$CDF = CDF_{\text{LARGE LOCA}} + CDF_{\text{SMALL LOCA}}$$

where

CDF = core damage frequency contribution from through-wall cracks

$CDF_{\text{LARGE LOCA}}$ = core damage frequency due to large LOCAs

$CDF_{\text{SMALL LOCA}}$ = core damage frequency due to small LOCAs .

The core damage frequency for a large LOCA is calculated as follows:

$$CDF_{\text{LARGE LOCA}} = F_{\text{TWC}} \bullet FRAC_{\text{LARGE LOCA}} \bullet COND\_CDF_{\text{LARGE LOCA}}$$

where

F_{TWC} = frequency of through-wall cracks (per year)

$FRAC_{\text{LARGE LOCA}}$ = fraction of through-wall cracks that result in large LOCA

$COND\_CDF_{\text{LARGE LOCA}}$ = conditional core damage probability given a large LOCA.

Similarly the core damage frequency for a small LOCA is calculated as:

$$CDF_{\text{SMALL LOCA}} = F_{\text{TWC}} \bullet FRAC_{\text{SMALL LOCA}} \bullet COND\_CDF_{\text{SMALL LOCA}}$$

where

F_{TWC} = frequency of through wall cracks (per year)

$FRAC_{\text{SMALL LOCA}}$ = fraction of through-wall cracks that result in small LOCA

$COND\_CDF_{\text{SMALL LOCA}}$ = conditional core damage probability given a small LOCA.

Plants and Components Considered

The plants and components (five PWR plants and two BWR plants) considered in this study are listed in Table 6. Stress amplitudes and numbers of stress cycles for the selected components during a 40-year plant life were taken from NUREG/CR-6260. The fracture mechanics calculations were based on stress tabulations from design stress reports. NUREG/CR-6260 did not reveal the identities of these plants.

Table 6 Plants and Components Considered in the Fatigue Study

PLANT TYPE	COMPONENT	LOCATION	MATERIAL
COMBUSTION ENGINEERING - NEW VINTAGE	REACTOR VESSEL	LOWER HEAD/SHELL	SA-533 GRADE B, CLASS 1
COMBUSTION ENGINEERING - NEW VINTAGE	REACTOR VESSEL	INLET NOZZLE	SA-508, CLASS 2
COMBUSTION ENGINEERING - NEW VINTAGE	REACTOR VESSEL	OUTLET NOZZLE	SA-508, CLASS 2
COMBUSTION ENGINEERING - NEW VINTAGE	SURGE LINE	ELBOW	SA-376, TYPE 316
COMBUSTION ENGINEERING - NEW VINTAGE	CHARGING NOZZLE	NOZZLE	SA-182, GRADE F1
COMBUSTION ENGINEERING - NEW VINTAGE	CHARGING NOZZLE	SAFE END	SA-182, TYPE 316
COMBUSTION ENGINEERING - NEW VINTAGE	SAFETY INJECTION NOZZLE	NOZZLE	SA-182, GRADE F1
COMBUSTION ENGINEERING - NEW VINTAGE	SAFETY INJECTION NOZZLE	SAFE END	SA-531, GRADE CF8M, TYPE 316
COMBUSTION ENGINEERING - NEW VINTAGE	SHUTDOWN COOLING LINE	ELBOW	SA-376, TYPE 316
COMBUSTION ENGINEERING - OLD VINTAGE	REACTOR VESSEL	AT LOWER HEAD TO SHELL JUNCTURE	SA-533, GRADE B, CLASS 1
COMBUSTION ENGINEERING - OLD VINTAGE	REACTOR VESSEL	INLET NOZZLE	SA-336
COMBUSTION ENGINEERING - OLD VINTAGE	REACTOR VESSEL	OUTLET NOZZLE	SA-336
COMBUSTION ENGINEERING - OLD VINTAGE	SURGE LINE ELBOW	ELBOW	SA-376, TYPE 316
COMBUSTION ENGINEERING - OLD VINTAGE	CHARGING NOZZLE	NOZZLE	SA-351, TYPE 316
COMBUSTION ENGINEERING - OLD VINTAGE	SAFETY INJECTION NOZZLE	NOZZLE	SA-351, TYPE 316
COMBUSTION ENGINEERING - OLD VINTAGE	SHUTDOWN COOLING LINE	ELBOW	SA-376, TYPE 316
BABCOCK AND WILCOX	REACTOR VESSEL	NEAR SUPPORT SKIRT JUNCTURE	SA-302, GRADE B
BABCOCK AND WILCOX	REACTOR VESSEL	OUTLET NOZZLE	SA-508, CLASS2
BABCOCK AND WILCOX	MAKEUP/HPI NOZZLE	SAFE END	SA-376, TYPE 316
BABCOCK AND WILCOX	DECAY HEAT REMOVAL LINE	REDUCING TEE	SA-376, TYPE 316
WESTINGHOUSE -NEW VINTAGE	REACTOR VESSEL	AT LOWER HEAD TO SHELL JUNCTURE	SA-533, GRADE B, CLASS 1
WESTINGHOUSE -NEW VINTAGE	REACTOR VESSEL	INLET NOZZLE	SA-508, CLASS 2
WESTINGHOUSE -NEW VINTAGE	REACTOR VESSEL	OUTLET NOZZLE	SA-508, CLASS 2
WESTINGHOUSE -NEW VINTAGE	CHARGING NOZZLE	NOZZLE	SA-182, F316N
WESTINGHOUSE -NEW VINTAGE	SAFETY INJEC NOZZLE	NOZZLE	SA-182, F316N
WESTINGHOUSE -NEW VINTAGE	RESIDUAL HEAT REMOVAL LINE	INLET TRANSITION	SA-376, TYPE 316
WESTINGHOUSE -OLD VINTAGE	REACTOR VESSEL	AT CORE SURPPORT GUIDE	SA-302, GRADE B
WESTINGHOUSE -OLD VINTAGE	REACTOR VESSEL	INLET NOZZLE INNER SURFACE	SA-302, GRADE B
WESTINGHOUSE -OLD VINTAGE	REACTOR VESSEL	INLET NOZZLE OUTER SURFACE	SA-302, GRADE B
WESTINGHOUSE -OLD VINTAGE	REACTOR VESSEL	OUTLET NOZZLE INNER SURFACE	SA-302, GRADE B
WESTINGHOUSE -OLD VINTAGE	REACTOR VESSEL	OUTLET NOZZLE OUTER SURFACE	SA-302, GRADE B
WESTINGHOUSE -OLD VINTAGE	CHARGING NOZZLE	NOZZLE	SA-182, TYPE 316
WESTINGHOUSE -OLD VINTAGE	SAFETY INJECTION NOZZLE	NOZZLE	SA-182, TYPE 316
WESTINGHOUSE -OLD VINTAGE	RESIDUAL HEAT REMOVAL LINE	TEE	SA-376, TYPE 316
GENERAL ELECTRIC-NEW VINTAGE	REACTOR VESSEL	NEAR CRDM PENETRATION	SA-508, CLASS 2
GENERAL ELECTRIC-NEW VINTAGE	FEEDWATER NOZZLE	SAFE END	SA-508, CLASS 1
GENERAL ELECTRIC-NEW VINTAGE	RECIRCULATION SYSTEM	TEE ON SUCTION PIPE	SA-358, TYPE 304
GENERAL ELECTRIC-NEW VINTAGE	CORE SPRAY LINE	SAFE END EXTENSION	SA-508, CLASS 1
GENERAL ELECTRIC-NEW VINTAGE	RESIDUAL HEAT REMOVAL LINE	STRAIGHT PIPE	SA-333, GRADE 6
GENERAL ELECTRIC-NEW VINTAGE	FEEDWATER LINE	ELBOW	SA-333, GRADE 6
GENERAL ELECTRIC-OLD VINTAGE	REACTOR VESSEL	AT LOWER HEAD TO SHELL TRANSITION	SA-302, GRADE B
GENERAL ELECTRIC-OLD VINTAGE	FEEDWATER NOZZLE	BORE	SA-508
GENERAL ELECTRIC-OLD VINTAGE	RECIRCULATION SYSTEM	RHR RETURN LINE	SA-358, TYPE 304, CLASS 1
GENERAL ELECTRIC-OLD VINTAGE	CORE SPRAY SYSTEM	NOZZLE	SA-302, GRADE B
GENERAL ELECTRIC-OLD VINTAGE	CORE SPRAY SYSTEM	SAFE END	SA-358, TYPE 304
GENERAL ELECTRIC-OLD VINTAGE	RESIDUAL HEAT REMOVAL LINE	TAPERED TRANSITION	SA-358, TYPE 304, CLASS 1
GENERAL ELECTRIC-OLD VINTAGE	FEEDWATER LINE	RCIC TEE	SA-106, GRADE B

The types and numbers of transients for the additional operating period corresponding to the 60-year plant life were assumed to be the same as those for the 40-year operating period. The number of cycles to crack initiation was a function of the material type, water/air environment, temperature, dissolved oxygen content, sulfur content and strain rate. The material types were carbon steel, low-alloy steel, 304/316 austenitic stainless steel and 316NG stainless steel. The statistical models of NUREG/CR-6335 were used to calculate the number of cycles to crack initiation corresponding to material S-N curves with given probabilities (or percentiles). For the PWR plants, the curves for high-sulfur steel (0.015 weight percent) and a low-oxygen environment (0.01-ppm) were used. For the BWR plants, the curves for high sulfur steel and a high-oxygen environment (0.10- ppm) were used. The strain rates for both PWR and

BWR components (low alloy and carbon steel) were assumed to be 0.001% (see NUREG/CR-6260). For 316 stainless steel, the strain rate was 0.004%. For all components, the operating temperature was assumed to be 290 C. The values of elastic modulus for carbon steels, low-alloy steels, and austenitic stainless steels were 27.0 ksi, 26.7 ksi, and 25.5 ksi, respectively.

Results of Calculations for Selected Components

Calculations were performed with pc-PRAISE to address the 47 selected components from the seven plants as listed in Table 6. These calculations predicted probabilities of crack initiation and probabilities of through-wall cracks as a function of time for plant operating periods up to 60 years. Probabilities of crack initiation were calculated using the fatigue life correlations from the ANL studies. The alternating stresses and anticipated number of cycles were the same as those used for the calculations of fatigue usage factors of NUREG/CR-6260.

The pc-PRAISE code was applied in the calculations, which implied that all the components were approximated by the cylindrical geometry of pipe with a circumferential crack. In many cases this geometry corresponded to the actual configuration of the component (safe ends). In other cases (nozzles and elbows) the pc-PRAISE model only approximated the actual component. Because crack initiation depends only on the peak local stresses, the actual component geometry was not a factor for initiation. To model crack initiation at multiple sites, the model required an input of the number of potential (2-inch long) initiation sites. A nominal diameter enabled the number of initiation sites to be estimated.

The Monte Carlo calculations with pc-PRAISE were run to a maximum of 10^6 simulations. Because some components had very low failure probabilities, this number of simulations was sometimes inadequate to establish probabilities of through-wall cracks. Additional calculations were then performed with a Latin Hypercube approach (Khaleel and Simonen 1998). The italicized values in Table 7 were derived from these supplementary calculations.

Table 7 provides a summary of the final results for all the components and gives failure probabilities for both a 40-year and a 60-year operating period. Many of the components have cumulative probabilities of crack initiation and through-wall cracks that approach unity. Other components, often with similar values of fatigue usage factors, show much lower failure probabilities. These preliminary results show maximum failure rates (through-wall cracks per year) of about $5x10^{-2}$, and maximum core damage of about $1.0x10^{-6}$ per year. These maximum values correspond to components with very high cumulative failure probabilities, for which the failure rates do not change significantly from 40 years to 60 years. Failure rates for other components having much lower failure probabilities are seen to increase by as much as an order of magnitude from 40 years to 60 years. These components however make relatively small overall contributions to core damage frequencies.

Table 7 Summary of Results for All Seven Plants

PLANT	COMPONENT	MAT	DET USEAGE@40	DET USEAGE@6	CUMULATIVE PI@40 YR	CUMULATIVE PI@60 YR	CUMULATIVE PTWC@40 YR	CUMULATIVE PTWC@60 YR	TWC/YEAR @40 YR	TWC/YEAR @60 YR	CDF @40 YR	CDF @60 YR
CE-NEW	RPV LOWER HEAD/SHELL	LAS	1.40E-02	2.10E-02	7.89E-06	4.82E-05	6.71E-16	1.44E-12	1.13E-15	1.90E-13	1.13E-16	1.91E-14
CE-NEW	RPV INLET NOZZLE	LAS	4.75E-01	7.12E-01	1.40E-02	4.44E-02	5.90E-05	9.01E-04	7.50E-06	7.59E-05	2.03E-11	2.05E-10
CE-NEW	RPV OUTLET NOZZLE	LAS	4.72E-01	7.08E-01	4.22E-01	6.89E-01	1.74E-03	2.90E-02	3.58E-04	2.57E-03	9.65E-10	6.93E-09
CE-NEW	SURGE LINE ELBOW	304/316	2.60E+00	3.90E+00	9.95E-01	9.99E-01	9.81E-01	9.98E-01	7.60E-02	9.38E-02	2.17E-06	2.67E-06
CE-NEW	CHARGING NOZZLE NOZZLE	LAS	1.04E-01	1.56E-01	9.56E-04	3.84E-03	2.61E-06	5.50E-05	3.46E-07	5.06E-06	2.77E-12	4.05E-11
CE-NEW	CHARGING NOZZLE SAFE END	304/316	5.02E-01	7.53E-01	1.06E-02	6.75E-02	9.00E-05	1.03E-03	1.75E-05	1.15E-04	1.40E-09	9.21E-09
CE-NEW	SAFETY INJECTION NOZZLE NOZZLE	LAS	4.57E-01	6.85E-01	1.01E-03	4.81E-03	1.00E-06	1.90E-05	3.75E-07	1.50E-06	1.88E-12	7.50E-12
CE-NEW	SAFETY INJECTION NOZZLE SAFE E	304/316	2.86E-01	4.29E-01	8.68E-03	3.16E-02	2.61E-06	5.50E-05	3.46E-07	5.06E-06	1.73E-11	2.53E-10
CE-NEW	SHUTDOWN COOLING LINE ELBOW	304/316	4.87E-01	7.30E-01	1.13E-02	5.75E-02	2.00E-05	4.53E-04	7.00E-08	4.40E-05	1.89E-10	1.19E-09
CE-OLD	RPV LOWER HEAD/SHELL	0	1.30E-02	1.95E-02	2.68E-06	1.93E-05	6.36E-16	1.85E-13	1.07E-16	1.85E-13	1.08E-17	1.86E-14
CE-OLD	RPV INLET NOZZLE	LAS	1.72E-01	2.58E-01	1.88E-03	7.89E-03	4.11E-07	1.33E-05	5.87E-08	1.33E-05	1.58E-13	3.59E-11
CE-OLD	RPV OUTLET NOZZLE	LAS	5.53E-01	8.29E-01	5.91E-01	8.46E-01	7.05E-02	3.53E-01	8.98E-03	2.27E-02	2.42E-08	6.13E-08
CE-OLD	SURGE LINE ELBOW	304/316	6.61E-01	9.92E-01	9.39E-01	9.87E-01	6.27E-01	8.85E-01	4.36E-02	5.48E-02	1.24E-06	1.56E-06
CE-OLD	CHARGING NOZZLE SAFE END	304/316	5.62E-01	8.43E-01	1.18E-02	5.31E-02	4.10E-05	5.98E-04	8.75E-06	5.05E-05	7.00E-10	4.04E-09
CE-OLD	SAFETY INJECTION NOZZLE SAFE E	304/316	3.17E-01	4.75E-01	7.56E-03	3.59E-02	1.40E-05	2.00E-04	2.25E-06	1.85E-05	1.13E-10	9.25E-10
CE-OLD	SHUTDOWN COOLING LINE ELBOW	304/316	8.40E-02	1.26E-01	3.94E-02	1.19E-01	2.10E-04	2.36E-03	4.38E-05	1.98E-04	1.18E-09	5.35E-09
B&W	RPV NEAR SUPPORT SKIRT	LAS	2.23E-01	3.35E-01	8.25E-03	2.50E-02	7.85E-06	1.52E-04	1.04E-06	1.36E-05	1.04E-07	1.36E-06
B&W	RPV OUTLET NOZZLE	LAS	4.69E-01	7.04E-01	7.74E-01	8.99E-01	1.83E-01	5.44E-01	1.94E-02	3.35E-02	5.25E-08	9.03E-08
B&W	MAKEUP/HPI NOZZLE SAFE END	304/316	1.05E+00	1.58E+00	1.30E-01	4.79E-01	2.10E-03	3.09E-02	5.88E-04	2.22E-03	2.94E-08	1.11E-07
B&W	DECAY HEAT REMOVAL/REDUCING T	304/316	5.30E-01	7.95E-01	5.72E-02	2.08E-01	3.00E-03	2.54E-02	4.26E-04	1.79E-03	1.15E-08	4.82E-08
W-NEW	RPV LOWER HEAD/SHELL	LAS	1.80E-02	2.70E-02	3.21E-05	1.71E-04	7.52E-13	9.64E-11	1.23E-13	1.21E-11	1.24E-14	1.50E-12
W-NEW	RPV INLET NOZZLE	LAS	2.90E-01	4.35E-01	2.49E-03	1.05E-02	9.17E-07	2.84E-05	1.30E-07	2.83E-06	3.51E-13	7.64E-12
W-NEW	RPV OUTLET NOZZLE	LAS	6.58E-01	9.87E-01	8.62E-01	9.49E-01	3.65E-01	7.42E-01	3.17E-02	4.50E-02	8.57E-08	1.22E-07
W-NEW	CHARGING NOZZLE NOZZLE	316NG	3.37E+00	5.06E+00	9.51E-01	9.83E-01	8.72E-01	9.63E-01	5.38E-02	5.66E-02	4.31E-07	4.53E-07
W-NEW	SAFETY INJEC NOZZLE NOZZLE	316NG	1.46E+00	2.19E+00	4.34E-03	3.69E-02	5.00E-04	1.09E-02	5.33E-05	1.30E-03	2.67E-10	6.50E-09
W-NEW	RESIDUAL HEAT INLET TRAN	304/316	2.73E+00	4.10E+00	9.58E-01	9.99E-01	7.80E-01	9.80E-01	6.25E-02	1.18E-01	1.69E-06	3.17E-06
W-OLD	RPV LOWER HEAD SHELL	LAS	8.91E-01	1.34E+00	1.11E-01	1.28E-01	7.20E-07	1.11E-05	8.38E-08	9.08E-07	8.44E-09	9.15E-08
W-OLD	RPV INLET NOZZLE INNER SURFACE	LAS	3.02E-01	4.53E-01	3.91E-01	6.44E-01	4.38E-03	5.04E-02	7.53E-04	3.98E-03	2.03E-09	1.07E-08
W-OLD	RPV INLET NOZZLE OUTER SURFACE	LAS	4.96E-01	7.44E-01	6.81E-02	1.11E-01	4.48E-04	3.32E-03	4.75E-05	2.18E-04	1.28E-10	5.89E-10
W-OLD	RPV OUTLET NOZZLE INNER SURF	LAS	4.99E-01	7.48E-01	4.90E-01	7.53E-01	9.33E-03	9.60E-02	1.56E-03	7.54E-03	4.21E-09	2.04E-08
W-OLD	RPV OUTLET NOZZLE OUTER SURF	LAS	3.47E-01	5.20E-01	1.63E-01	2.38E-01	7.77E-03	3.60E-02	6.99E-04	1.83E-03	1.89E-09	4.94E-09
W-OLD	CHARGING NOZZLE NOZZLE	304/316	3.19E-01	4.79E-01	4.67E-04	3.75E-03	3.00E-07	5.20E-06	7.50E-08	6.00E-07	6.00E-13	4.80E-12
W-OLD	SAFETY INJECTION NOZZLE NOZZLE	304/316	3.27E-01	4.90E-01	1.88E-03	1.31E-02	4.00E-06	8.80E-05	8.75E-07	1.05E-05	4.38E-12	5.25E-11
W-OLD	RESIDUAL HEAT REMOVAL TEE	304/316	2.05E-01	3.08E-01	1.34E-02	5.16E-02	1.15E-04	1.14E-03	1.63E-05	9.26E-05	8.13E-10	4.63E-09
GE-NEW	RPV NEAR CRDM PENETRATION	LAS	6.28E-01	9.42E-01	7.89E-05	3.49E-04	7.88E-12	6.82E-10	1.25E-12	8.26E-11	1.25E-13	8.26E-12
GE-NEW	FEEDWATER NOZZLE SAFE END	LAS	1.88E+00	2.82E+00	1.04E-01	2.53E-01	1.31E-03	1.47E-02	2.38E-04	1.23E-03	3.57E-11	1.84E-10
GE-NEW	RECIRC SYS - TEE SUCTION PIPE	304/316	8.30E-01	1.25E+00	4.23E-02	1.39E-01	4.80E-04	4.67E-03	7.13E-05	3.66E-04	1.07E-11	5.49E-11
GE-NEW	CORE SPRAY LINE SAFE END EXT	LAS	4.36E-01	6.54E-01	3.83E-04	1.27E-03	1.45E-07	3.25E-06	1.97E-08	3.04E-07	7.09E-15	1.09E-13
GE-NEW	RHR LINE STRAIGHT PIPE	LAS	1.13E+01	1.69E+01	4.73E-01	6.71E-01	4.10E-01	6.21E-01	1.35E-02	2.25E-02	2.54E-11	2.03E-10
GE-NEW	FEEDWATER LINE ELBOW	LAS	3.69E+00	5.53E+00	1.59E-01	3.65E-01	1.01E-03	1.46E-02	1.69E-04	1.35E-03	3.04E-09	5.06E-09
GE-OLD	RPV LOWER HEAD TO SHELL	LAS	7.90E-02	1.19E-01	2.71E-10	2.76E-08	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
GE-OLD	RPV FEEDWATER NOZZLE BORE	LAS	3.17E+00	4.75E+00	7.27E-02	2.42E-01	1.00E-05	8.80E-04	2.50E-06	9.76E-05	3.75E-14	1.46E-12
GE-OLD	RECIRC SYSTEM RHR RETURN LINE	304/316	3.90E+00	5.85E+00	9.43E-01	9.99E-01	7.12E-01	9.85E-01	7.20E-02	1.23E-01	1.08E-08	1.84E-08
GE-OLD	CORE SPRAY SYSTEM NOZZLE	LAS	5.20E-01	7.80E-01	1.41E-04	7.89E-04	1.91E-08	8.84E-07	2.85E-09	9.51E-08	6.41E-17	2.14E-15
GE-OLD	CORE SPRAY SYSTEM SAFE END	304/316	1.77E+00	2.65E+00	3.33E-01	7.64E-01	1.46E-02	1.10E-01	2.08E-03	8.04E-03	4.68E-10	1.81E-09
GE-OLD	RESIDUAL HEAT TAPERED	304/316	4.78E-01	7.17E-01	1.47E-03	7.89E-03	9.21E-05	1.02E-03	1.07E-05	7.82E-05	1.61E-12	1.17E-11
GE-OLD	FEEDWATER LINE - RCIC TEE	LAS	6.98E+00	1.05E+01	3.86E-01	7.82E-01	2.99E-03	5.92E-02	6.96E-04	5.54E-03	1.04E-10	8.30E-10

Through-Wall Crack Probabilities Versus Usage Factors - Figure 6 shows the correlation between calculated probabilities of through-wall cracks with the fatigue usage factors reported in NUREG/CR-6260. The relatively poor correlation is related to the fact that fatigue usage factors address only crack initiation, and do not address the specific factors that determine if an initiated crack will grow to become a through-wall crack. The plot does indicate that fatigue failures can be expected (probability of failure greater than $>10^{-1}$) even for usage factors less than one. Usage factors greater than unity can result in essentially 100 percent failure probability. Figure 6 also indicates that the probabilities of failure become relatively low (10^{-3} or less) for usage factors of 0.1 or less. The predictions over all trends of Figure 6 are consistent with the position that code usage factors are not intended to be precise predictions of cycles to fatigue failure, but rather provide a method to establish acceptable designs. It is important to note that plant operating experience has shown few if any fatigue failures for the loading conditions identified in the design calculations. Instead, fatigue failures have generally been due to stresses (vibration, thermal fatigue, etc.) which were not anticipated during plant design.

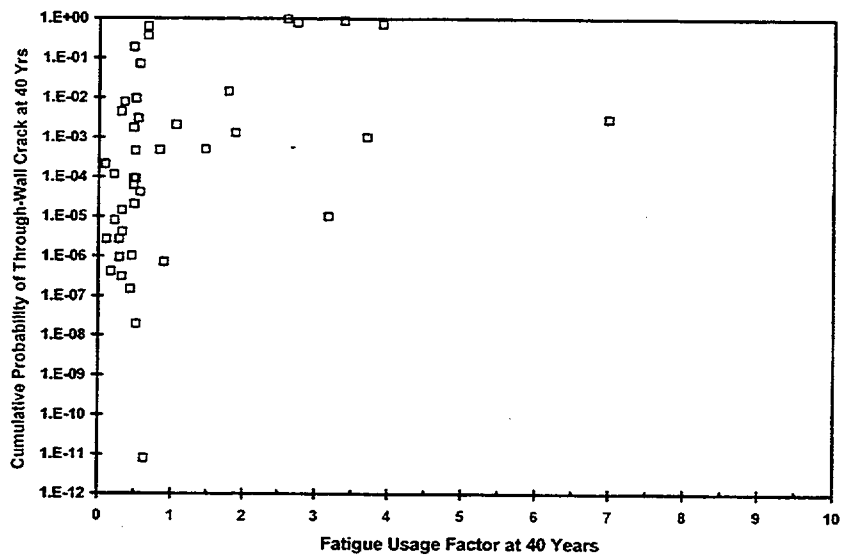

Figure 6 Comparison of Calculated Usage Factors with Calculated Through-Wall Crack Probabilities

Probabilities at 60-Years Versus 40-Years - Figures 7–10 show trends of the data from Table 7. The plots display both the overall range of the data, and compare the probabilities at the end of a 60-year plant life with the corresponding probabilities for a 40-year plant life. The range of the through-wall crack probabilities (Figure 8) covers about seven orders of magnitude. The component-to-component range for core damage frequencies (Figure 10) is almost 12 orders of magnitude.

The probabilities and frequencies for the 60-year plant life can be a factor of 10 or greater than for the 40-year plant life. These relative differences are greatest for those components that have relatively small failure probabilities at 40 years. In contrast, there are only small increases between 40 to 60 years when the 40-year probabilities are already quite large. The through-wall crack frequencies (Figure 9) saturate at a value of about 5×10^{-2} (through-wall cracks per year per component), with little increase between 40 and 60 years. Such components have cumulative failure probabilities that approach unity even at a plant life of 40 years. In these cases the fatigue cracks initiate relatively early in life resulting in a high potential of leaking during the 40-year operating period. It is unlikely that such high failure probabilities could occur except at a few isolated fatigue sites. Early failures would prompt corrective action programs consisting of augmented inspections, repairs and replacements, and changes to plant operating practices. The present model neglects the significant reductions in failure frequencies from such programs.

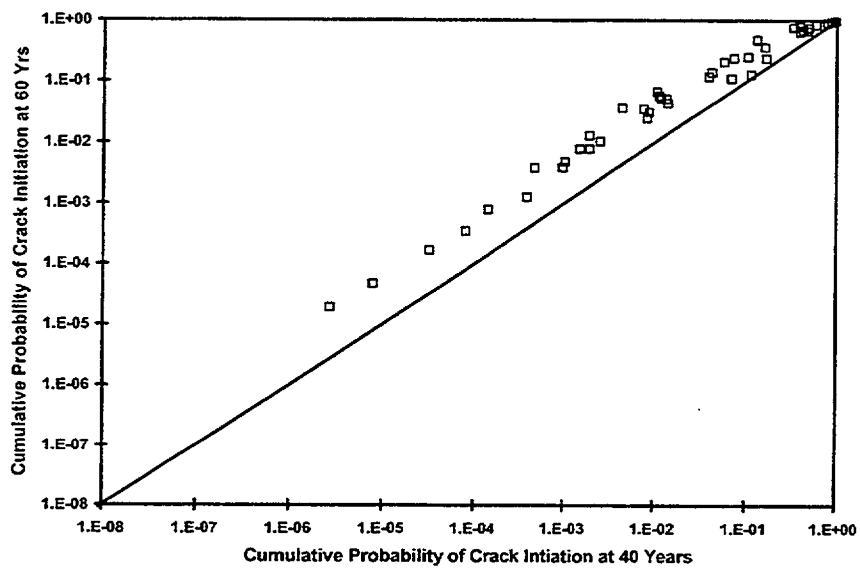

Figure 7 Cumulative Probability of Crack Initiation at 40 Years Versus 60 Years

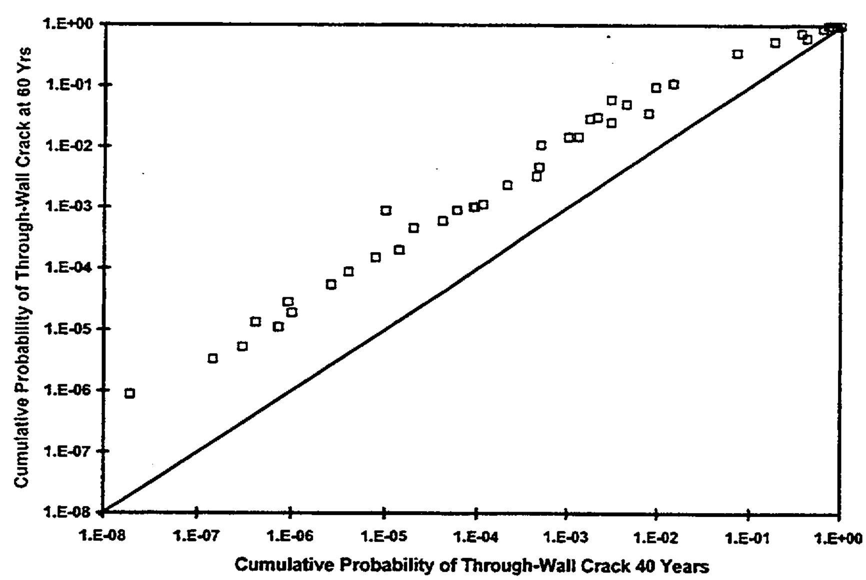

Figure 8 Cumulative Probability of Through-Wall Crack at 40 Years Versus 60 Years

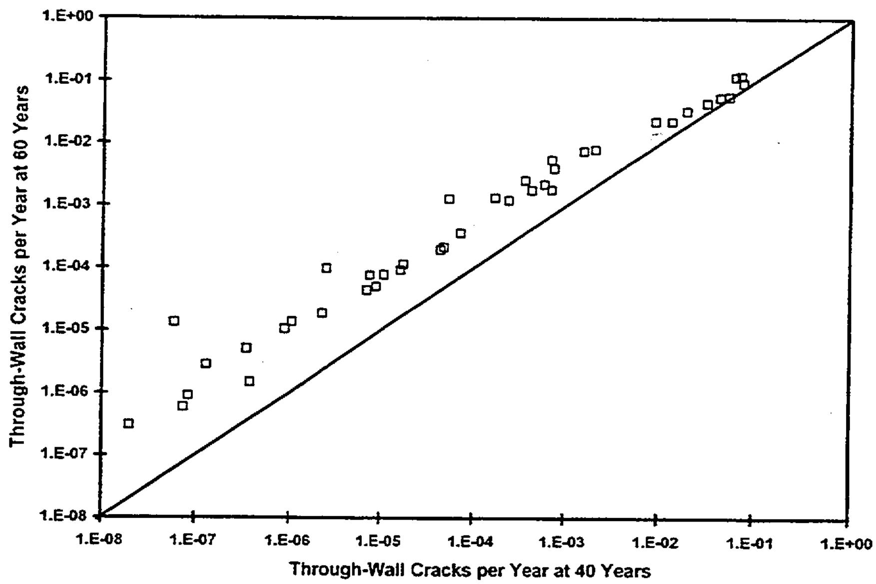

Figure 9 Through-Wall Cracks per Year at 40 Years Versus 60 Years

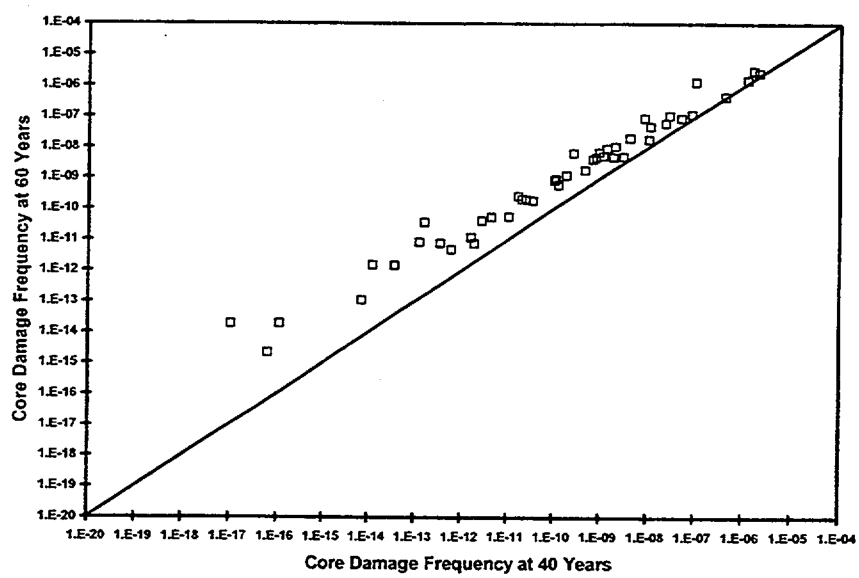

Figure 10 Core Damage Frequency at 40 Years Versus 60 Years

Only three of the components in Table 7 have calculated core damage frequencies (CDF) that exceed 10^{-6} per year. It is also noted that the core damage frequencies for the BWR type plants are generally much lower than those for the PWR plants. This is a direct result of the lower probabilities of core damage given the occurrence of small and large leaks. The two locations of most concern (CDF > 1.0E-06) are the surge line elbow in the newer vintage Combustion Engineering plant and the residual heat removal system inlet transition for the newer vintage Westinghouse plant.

Water Versus Air Environment - Additional calculations were performed to establish the separate effects of reactor coolant environment independent of the issue of 40 years versus 60 years. Figures 11 and 12 use the 40-year life for the water environment as the baseline case. The data show that changing to an air environment gives lower probabilities of both crack initiation and through-wall cracks by about a factor of 100. Changing from a 40-year life to a 60-year life increases the probabilities by a factor of about 10. This increase is significantly less than the effect associated with the environmental effects.

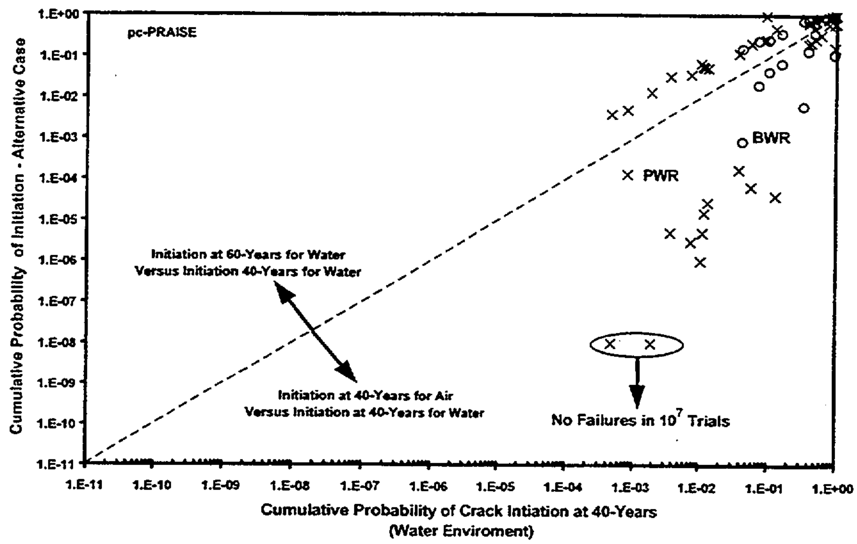

Figure 11 Comparison of Probabilities of Fatigue Crack Initiation for Air Versus Water Environment and for 40-Year Life and 60-Year Life.

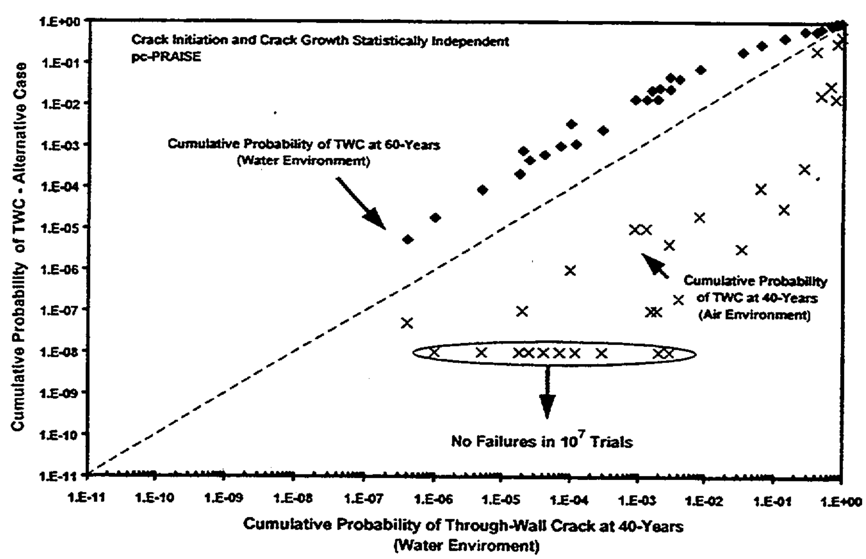

Figure 12 Comparison of Probabilities of Through-Wall Cracks for Air Versus Water Environment and for 40-Year Life and 60-Year Life.

Summary and Conclusions

Probabilities of fatigue failures and the associated core damage frequencies have been estimated for RPV and piping components of five PWR and two BWR plants. These calculations were made possible by the development of a new version of the pc-PRAISE probabilistic fracture mechanics code that has the ability to simulate the initiation of fatigue cracks in combination with a simulation of the subsequent growth of these fatigue cracks. The calculations gave a wide range of failure probabilities for the selected components, with some components having end-of-life probabilities of through-wall crack of nearly 100 percent and others with probabilities less than 10^{-6}. It is recognized that there are uncertainties in these calculated failure probabilities. The results should therefore be considered interim and are subject to further review. Sources of the uncertainties come from assumptions made in the fracture mechanics model itself and from the inputs to the model. Uncertain inputs include the design-basis data for the cyclic stresses which could differ from the stresses occurring during the actual plant operation, and assumptions regarding strain rates and environmental variables used for the prediction of fatigue crack initiation. In spite of these uncertainties, the results of the calculations are useful when they are applied in terms of relative probabilities, such as for comparing through-wall crack frequencies at the end of a 40-year plant life to those at the end of a 60-year plant life.

The calculations indicate that the components with the very high probabilities of failure can have through-wall crack frequencies that approach about 5×10^{-2} per year. In contrast, other components with much lower failure probabilities can have their failure frequencies increase by factors of about 10 from 40 years to 60 years. Calculations were also performed to address the effects of reactor water environments (versus air) and to compare these effects to the effects of extended plant operation from 40 years to 60 years. The environmental effects were predicted to increase through-wall crack probabilities by as much as two orders of magnitude

Contributions to core damage frequencies have also been estimated in a conservative/bounding manner for each of the vessel and piping components. The maximum calculated contributions are on the order of 10^{-6} per year. This number is subject to uncertainties that were addressed in a conservative manner by assigning conservative values for 1) conditional probabilities of small and large LOCAs, and 2) probabilities of core damage given the occurrence of these failure modes. The results indicate that the calculated core damage frequencies for the components with the highest calculated frequencies of failure show essentially no increase from 40 to 60 years.

References

Bush, S.H., Do, M.J., Slavish A.L. and Chockie, A.D. 1996. "Piping Failures in United States Nuclear Power Plants: 1961 – 1995," SKI Report 96:20, prepared for Swedish Nuclear Power Inspectorate.

Chopra, O., 1998. "Updated Fatigue Design Curves for Austenitic Stainless Steel in LWR Environments," Private Communication.

Forman, R.G., Y. Shivakumar, J.C. Newman Jr., S.M. Piotrowski, and L.C. Williams. 1988. "Development of the NASA/FLAGRO Computer Program," in *Fracture Mechanics: Eighth Symposium*, ASTM STP 945, Eds. D.T. Read and R.P. Read, pp. 781-804, American Society for Testing and Materials, Philadelphia.

Harris, D.O., E.Y. Lim, and D.D. Dedhia. 1981. "Probability of Pipe Fracture in the Primary Coolant Loop of a PWR Plant – Volume 5: Probabilistic Fracture Mechanics Analysis," NUREG/CR-2189, Vol. 5, U.S. Nuclear Regulatory Commission.

Harris, D.O. and D.D. Dedhia. 1992. "A Probabilistic Fracture Mechanics Code for Piping Reliability Analysis (pc-PRAISE code)," NUREG/CR-5864.

Keisler, J., O.K. Chopra, and W.J. Shack. 1994. "Statistical Analysis of Fatigue Strain-Life Data for Carbon and Low-Alloy Steels," NUREG/CR-6237.

Keisler, J., Chopra, O.K. and Shack, W.J., 1995. "Fatigue Strain-Life Behavior of Carbon and Low-Alloy Steels, Austenitic Stainless Steels, and Alloy 600 in LWR Environments," NUREG/CR-6335.

Khaleel, M.A. and F.A. Simonen. 1998. "A Probabilistic Model for Fatigue Crack Initiation and Propagation," *Fatigue, Fracture and Residual Stresses*, PVP-373, pp. 27-34, American Society of Mechanical Engineers, New York.

Memoradum dated September 23, 1994, from E.S. Beckjord to A.C. Thadani, "Generic Issue 78, Monitoring of Fatigue Transient Limits for the Reactor Coolant System".

Mjumdar, S., O.K. Chopra, and W.J. Shack. 1993. "Interim Fatigue Curves to Selected Nuclear Power Plant Components," NUREG/CR-5999.

Simonen, F.A., D.O. Harris, and D.D. Dedhia. 1998. "Effect of Leak Detection on Piping Failure Probabilities," *Fatigue, Fracture and Residual Stresses*, PVP-373, pp. 105-113, American Society of Mechanical Engineers, New York.

Ware, A.G., D.K. Morton, and M.E. Nitzel, 1995. "Application of NUREG/CR-5999 Interim Fatigue Curves to Selected Nuclear Power Plant Components," NUREG/CR-6260.

Westinghouse Owners Group. 1997. "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," WCAP-14572 Revision 1.

Wire, G.L. and Y.Y. Li. 1996. "Initiation of Environmentally-Assisted Cracking in Low Alloy Steels," *Fatigue and Fracture – Volume 1*, PVP-Vol.323, pp. 269-289, American Society of Mechanical Engineers, New York.

Severe Accident Risks: An Assessment of Five U.S. Nuclear Power Plants, NUREG-1150, Summary Report , Second Draft for Peer Review, U.S. Nuclear Regulatory Commission, Washington, D.C.

Structural Evaluation of Electrosleeved Tubes Under Severe Accident Transients

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Abstract

A flow stress model was developed for predicting failure of Electrosleeved PWR steam generator tubing under severe accident transients. The electrosleeve, which is nanocrystalline pure nickel, loses its strength at temperatures greater than 400°C during severe accidents because of grain growth. A grain growth model and the Hall-Petch relationship were used to calculate the loss of flow stress as a function of time and temperature during the accident. Available tensile test data as well as high temperature failure tests on notched Electrosleeved tube specimens were used to derive the basic parameters of the failure model. The model was used to predict the failure temperatures of Electrosleeved tubes with axial cracks in the parent tube during postulated severe accident transients.

1. Introduction

Prediction of failure of a complex composite material like the Electrosleeved steam generator tubing under severe accident transients is a difficult problem. The electrosleeve material is almost pure Ni and derives its strength and other useful properties from its nanocrystalline microstructure, which is stable at reactor operating temperatures. However, it undergoes rapid grain growth at the high temperatures that are expected during severe accidents, resulting in a loss of strength and a corresponding decrease in the flow stress. The magnitude of this decrease depends on the time-temperature history during the accident. Low-temperature tensile data on the electrosleeve material (without the tube) are available, but the available tensile data on electrosleeve at high temperatures in either the aged or unaged condition are very limited. Initially, the assumption was made that there were no experimental failure data available on Electrosleeved tube with cracks. Therefore, analytical models were exclusively relied upon to predict failure of the composite structure with cracks, using the available tensile data to determine the parameters of the models.

Following initial modeling and analysis, Framatome Technologies, Inc. (FTI) provided failure data from six tests on unsleeved and Electrosleeved tubes with and without notches under simulated severe accident loading. In contrast to the prediction by the model that the damaging effect of notch length should level off with increasing notch length, the FTI test data indicated that the failure temperature of the Electrosleeved tube decreased almost linearly with notch length. On the other hand, the failure temperatures of the unsleeved and degraded Alloy 600 tubes were predicted reasonably well by the flow stress model presented in Ref. 1.

Subsequent to their initial tests, FTI supplied twelve Electrosleeved tube specimens, three of which had 2 in. long throughwall notches for testing at Argonne National Laboratory (ANL). Eight other specimens were notched by electro-discharge machining at ANL with notch lengths of 0.5, 1, and 3 in., all nominally 100% throughwall of the parent tube. All specimens were tested to failure using a temperature and pressure history that closely simulated those for a station blackout (SBO) with a depressurized secondary side (Case 6).[2] This paper describes the basis for the analytical models and shows how the basic parameters of the models were revised on the basis of the test results. Although the models overestimated the failure temperatures of the FTI tests by 35–70°C initially, the failure temperatures were predicted within ±15°C using the revised parameters. Finally, the models are used to predict failure of Electrosleeved steam generator tubing during postulated severe accidents. The reference geometry considered in this paper is a 7/8-in.–diameter, 0.050-in.-wall-thickness Alloy 600 tubing, with a nominal 0.038-in.-thick electrosleeve (Fig. 1).

2. Flow Stress Criterion

The flow stress failure criterion for tubes with axial cracks can be stated as follows[1]:

$$\sigma_{lig} = H(T,t), \tag{1a}$$

where H is the flow stress (dependent on the temperature history) of the electrosleeve and σ_{lig} is the ligament stress, given by Eq. 1b.

$$\sigma_{lig} = m_p \sigma_h \tag{1b}$$

where m_p, which depends on the axial crack length and depth, is the ligament stress magnification factor, σ_h is the nominal hoop stress (calculated using the mean radius and total thickness of the tube and the sleeve)

2.1 Determination of m_p for axial cracks

Initially the hoop stress magnification factor m_p for the crack tip ligament in the electrosleeve was estimated from the ANL equation for a single-layer shell used in Ref. 1. However, the m_p factor could be reduced if the flow stress of the electrosleeve ligament is significantly lower than that of the parent tube. In fact, detailed analyses of available tensile data of the electrosleeve (to be discussed later) showed that, at the temperatures of interest, the flow stress ratio between the parent tube and the electrosleeve varies between 2-3. To determine the effect of the flow stresses of the electrosleeve and Alloy 600 on m_p, a series of finite element analyses (FEA) was conducted for a bi-layer tube with a 100% throughwall crack in the outer layer (simulating Alloy 600) under a constant temperature and increasing pressure loading. The ratio between the flow stress of the outer layer and the inner layer (simulating the electrosleeve) was varied between 1 and 3. The results, plotted in Fig. 2a, confirm that the values of m_p are indeed reduced significantly when the flow stress ratio is increased. Note that the m_p value calculated by FEA for flow stress ratio =1 is close to that derived from the ANL correlation for cracks ≤ 1 in., but levels off with increasing crack length beyond 2 in. On the other hand, the m_p calculated by the ANL correlation continues to increase with increasing crack length, although the actual increases are small beyond a crack length of 2 in.

Because the FEA grid may not have been sufficiently fine to obtain highly accurate solutions, the FEA results were used to calculate the ratio between the m_p for the Electrosleeved tube and the homogeneous tube as a function of the ratio between the flow stress of the parent tube and the electrosleeve, as shown in Fig. 2b. This m_p ratio was then used to scale the m_p calculated by the ANL correlation for a homogeneous tube to obtain the effective m_p of the Electrosleeved tubes with notches as indicated in Eq. 2 (FSR denotes flow stress ratio):

$$m_p(\text{eff.}) = \frac{m_p(\text{FEA})}{m_p(\text{FEA}, \text{FSR}=1)} \times m_p(\text{ANL}) \qquad (2)$$

3. Material Properties Data for Electrosleeve

Data provided by FTI show that the electrosleeve material is stronger than the tube material at the reactor operating temperature. However, at high temperatures (≥400°C), the electrosleeve begins to lose its hardness because of grain growth (Fig. 3a). The thermal aging effect is a complicated phenomenon consisting of at least two steps. In the first step, the phosphide precipitates in the grain boundary, which prevent grain growth, are dissolved, and in the second step grain growth occurs. The starting or initial hardness of the FTI isothermal aging specimens show a very large specimen-to-specimen scatter. Therefore, the loss of hardness data for each specimen was normalized with respect to its initial hardness, as shown in Fig. 3a. The data in Fig. 3a suggest that the hardness of the material decreases with time, albeit at a relatively slow rate, starting very early. The nucleation times for this process for specimens aged at > 425°C are relatively short and are ignored in the nucleation model to be discussed later. The data in Fig. 3a also suggest the existence of a second process with longer nucleation times that involves very rapid decrease in hardness with time and is very likely linked to the process of rapid grain growth. The reciprocal of the incubation time for the onset of rapid loss of hardness (rapid grain growth) has a temperature-dependent activation energy as shown in Fig. 3b. For analyses of loss of hardness, the continuously varying activation energy curve Q was replaced by the step function indicated in Fig. 3b. Sensitivity studies showed that the results are not sensitive to the form chosen for Q.

Data for the yield and ultimate tensile strengths of the electrosleeve material were reported by FTI from room temperature to 343°C (650°F). The flow stress, which is the average of the yield and ultimate tensile strengths, as a function of temperature is shown in Fig. 4. The single high temperature electrosleeve data point in Fig. 4 was estimated from a tensile test conducted on aged material and will be discussed later. Initially, in the absence of any other flow stress data at high temperature, the solid line in Fig. 4 was used as an estimate for the unaged flow stress curve of the electrosleeve. It should be remembered that, during a severe accident, the actual flow stress of the electrosleeve is reduced from the unaged curve (H_i) shown in Fig. 4 because of grain growth. The high temperature tests conducted by ANL (to be discussed later) and FTI suggested that the unaged flow stress of the electrosleeve material is less than that shown in Fig. 4.

FTI also submitted to the NRC data from a series of tensile tests at 343°C on specimens that were exposed to isothermal pre-aging treatment at high temperatures for various times (Fig. 5a). The data at 760°C is for a tensile test that was conducted at 760°C.

FTI also provided failure data from six tests on internally pressurized tubes that were subjected to a variety of temperature ramps simulating those expected during SBO accidents. The initial temperature ramp rate up to ~ 500°C varied between 3- 5°C/min, which was generally followed by a ramp rate of 7-9°C/min until failure. However, in some cases the ramp rate was gradually decreased to 1.2 °C/min above 705°C. Three tests were conducted on unsleeved Alloy 600 tubes with and without any degradation and three on Electrosleeved tubes (7/8 in. Dia.) with 0.5, 1 and 2 in. throughwall axial notches in the parent tube.

4. Analytical Models

Two analytical models were originally developed for estimating the failure temperature under severe accident transients – a model based on linear damage rule[3] and a model based on the Hall-Petch relationship[4,5]. In both models, a basic assumption is the existence of a temperature-dependent unaged (i.e., without grain growth) flow stress curve of the electrosleeve. This unaged flow stress curve is largely a theoretical construct of the models because to establish it directly from tensile tests at high temperatures would be difficult due to the grain growth that would occur in the specimens unless the specimens could be heated up, stabilized, and tested very rapidly. Therefore, it was calculated from high temperature failure data using the models. Ideally, high temperature failure tests on specimens subjected to severe accident temperature and pressure ramps should be used to derive the flow stress curve of the electrosleeve. However, such test data were not available when the models were first developed. Therefore, the unaged flow stress curve of the electrosleeve was derived initially by analysis of a single tensile test on a specimen pre-aged and tested at 760°C (rectangle in Fig. 4) and the FTI tensile test data at ≤ 343°C (Fig. 4). Subsequently, the flow stress curve of the electrosleeve was revised on the basis of high temperature failure tests conducted at ANL.

Both models gave comparable results for failure temperatures. Since the Hall-Petch model was more mechanistically based, it was selected for use in failure prediction.

4.1 Model based on Hall-Petch equation

In this model the "nucleation" phase is explicitly separated from the "growth" phase of the grain growth phenomenon. As mentioned earlier, it was assumed that the electrosleeve has an initial "unaged" flow stress curve $H_i(T)$, e.g., Fig. 4. The hardness or flow stress (at a sufficiently high strain rate) of the electrosleeve material was assumed to depend on the grain size by the Hall-Petch relationship, i.e.,

$$H(T) = Ad^{-n}f(T), \tag{3}$$

where H(T) is the flow stress at any temperature T, d is the grain diameter, n is the Hall-Petch exponent, and f(T) is a correction factor for temperature. Although Eq. 3 may not be applicable to nanocrystalline material, it is assumed to provide reasonable estimates for flow stress at the failure temperature after the material has undergone significant grain growth. During high temperature exposure, the growth rate of grain diameter was assumed as Eq. 4.

$$\dot{d} = \begin{cases} 0 \text{ for } t < t_n \\ \dfrac{B}{d}\exp\left(\dfrac{-Q_g}{RT}\right) \text{ for } t \geq t_n, \end{cases} \tag{4}$$

where t_n is the nucleation time to loss of flow stress (i.e., onset of grain growth), B is a constant, Q_g is the activation energy for grain growth, R=1.987 cal/mole/°C. Recrystallization due to plastic straining was ignored. The form of the grain growth rate equation was chosen such that, under isothermal aging, the grain growth follows a parabolic law. Under isothermal aging, the reciprocal of the nucleation time ($1/t_n$), which has an activation energy Q_n, is given by the following equation:

$$\frac{1}{t_n} = C\exp\left(\frac{-Q_n}{RT}\right) \tag{5}$$

where C is a constant. The variation of Q_n with T is given in Fig. 3b.

The tensile data reported by FTI on pre-aged specimens of electrosleeve material were used to calculate the values of various parameters in Eqs 3-5. Integrating Eq. 4, using Eq. 5 and assuming $Q_n = Q_g = Q$,

$$d(t) = \left[d_i^2 - \frac{2B}{C} + 2Bt\exp\left(\frac{-Q}{RT}\right) \right]^{1/2}, \tag{6}$$

where d_i is the grain diameter of the as-received material and T is the aging temperature. Substituting Eq. 6 into Eq. 3, denoting the tensile testing temperature as T_o, the initial "unaged" flow stress at T_o as H_o and solving,

$$t\exp\left(\frac{-Q}{RT}\right) = \frac{d_i^2}{2B}\left[\left(\frac{H_o}{H}\right)^{2/n} - 1\right] + \frac{1}{C}, \tag{7a}$$

where

$$H_o = Ad_i^{-n}f(T_o). \tag{7b}$$

Results from the FTI tensile data (T_o=343°C) on pre-aged specimens are plotted in Fig. 5b using coordinates suggested by Eq. 7a with n =0.33. Values of $d_i^2/2B$, and 1/C were obtained from the slope and intercept of the linear fit. As mentioned earlier, the specimen that was aged for 30 min at 760°C was also tensile tested at 760°C. Since, prior to constant temperature aging, this specimen was ramped from 327°C to 760°C at the slow rate of 5.8°C/min, an analysis using an activation energy of 35 kcal/mole gave an effective aging time at 760°C of 39 min. A reduction factor for the flow stress at 760°C compared to that at 343°C was obtained by fitting the data, excluding the data at 760°C, by a best-fit line and extending the line to the value of the time-temperature parameter corresponding to the test at 760°C, as shown by dotted line in Fig. 5b. The rectangular symbol in the flow stress curve shown in Fig. 4 is the estimated value of unaged flow stresses at 760°C as obtained by applying the reduction factor to the flow stress at 343°C.

Nucleation times to onset of loss of flow stress (i.e., grain growth) under isothermal aging were calculated using Eq. 5 and the step-wise varying approximation to the activation energy data shown in Fig. 3b. The calculated nucleation times and those for the rapid loss of hardness as derived from the FTI nucleation data (Fig. 3a), plotted in Fig. 6a, show that n=0.33 fits the data better.

Under a variable temperature history, Eq. 5 can be generalized to give the time to nucleation as follows:

$$C\int_0^{t_n} \exp\left(\frac{-Q}{RT(t)}\right)dt = 1 \tag{8}$$

Similarly, Eq. 4 can be integrated to give the grain diameter at any time t.

$$d(t)=\begin{cases} d_i & \text{for } t<t_n \\ \left[d_i^2+2B\int_{t_n}^{t}\exp\left(\frac{-Q}{RT(t)}\right)dt\right]^{1/2} & \text{for } t\geq t_n \end{cases} \tag{9}$$

Substituting Eqs 8-9 into Eq. 3 and solving for the flow stress H at any time,

$$H(t)=\begin{cases} H_i(t) & \text{for } t<t_n \\ \left[1+\frac{2B}{d_i^2}\int_{t_n}^{t}\exp\left(\frac{-Q}{RT(t)}\right)dt\right]^{-n/2} H_i(t) & \text{for } t\geq t_n \end{cases} \tag{10}$$

where $H_i(t)$ is the initial "unaged" flow stress at T(t). Ligament failure is predicted to occur when Eq. 1a is satisfied.

In most cases, the nucleation times for onset of rapid loss of flow stress are negligible compared to the times to failure. As an example, the variation of flow stress of the electrosleeve (calculated by Eq. 10) subjected to a typical station blackout scenario[2] (SBO Case 6) is plotted in Fig. 6b for two values of the exponent n. Note that although the nucleation temperature for onset of loss of flow stress corresponding to the two n values are significantly different, the flow stress curves approach each other and are coincident above 620°C. Since most of the test failure temperatures (to be discussed later) are generally >620°C, any inaccuracy in the nucleation model should not influence the predicted failure temperatures significantly.

5. Predicted Versus Observed Failure temperatures

The studies in Ref. 2 showed that the most severe challenge to the integrity of steam generator tube arises from station blackout (SBO) sequences in which the secondary system dries out and the primary system fails to depressurize (a "high–dry" sequence, Case 6). In this case the Δp across the tube wall is approximately constant = 2.35 ksi and the time–temperature history is given in Fig. 7. For this case the tube temperature is 684°C when the surge line failure occurs.

5.1 Initial Failure Predictions for FTI tests

FTI simulation of the SBO (Case 6) temperature ramp consisted of a variety of temperature ramps with the initial rate (at < 500°C) varying between 3 and 5°C/min. Above a temperature of 500°C, the notched Electrosleeved specimens were ramped at 7-9°C/min to failure, except for the test with a 0.5 in. notch for which the ramp rate was gradually reduced to 1.2°C/min above 705°C.

The model using the Hall-Petch equation to represent the changes in the flow stress was used to predict failure temperatures of the tests conducted by FTI on 7/8 in. diameter unsleeved tubes, both degraded and undegraded, and Electrosleeved tubes with 100% deep axial notches in the parent tube. The details of the notch and tube geometry of the specimens are included in Table 1. Failure temperatures were calculated using the temperature ramps for each specimen supplied by FTI. The predicted failure temperatures for the notched unsleeved Alloy 600 tubes were independent of the temperature history. The comparison between the predicted and observed failure temperatures is shown in Table 2. The predictions for the Electrosleeved tubes in Table 2 were made using the high–temperature flow stress curves given in Fig. 4. The predicted failure temperatures overestimate the experimentally observed failure temperatures of the Electrosleeved tubes by 35-70°C. Note that these predictions were made without the benefit of a single high-temperature pressurized tube test. The predictions are in much better agreement with the observed values, if a modified flow stress curve that includes the results of high temperature pressurized tube tests, is used, as will be discussed later.

The failure temperatures of the two unsleeved degraded Alloy 600 tubes were predicted quite well by the flow stress model of Ref.1. Note that these two tests are consistent with each other because the m_p value for a 50% deep 2 in. crack is approximately 2, which is also the hoop stress magnification factor for a 50% uniformly thinned tube. The test on undegraded and unsleeved Alloy 600 involved a hold at constant temperature, which the flow stress model cannot handle. However, the creep rupture model presented in Ref. 1 can predict the failure time within a factor of 2.

5.2 ANL test results and revised unaged flow stress curve

As mentioned earlier, FTI provided twelve Electrosleeved specimens three of which were notched. Eight additional specimens were notched (≈ 0.0075 in. wide) at ANL by electrodischarge machining. Eleven tests were conducted at ANL. The time-temperature history for these tests closely simulated the SBO sequence identified as Case 6 in Ref. 2 and consisted of holding the pressure differential constant at 2.35 ksi while ramping the temperature from 300°C to 545°C at 4.2°C/min followed by 12.5°C/min ramp until failure. (solid line in Fig. 7). Note that the FTI tests were conducted using different temperature ramps, as discussed in section 5.1. A summary of all the tests conducted by ANL as well as by FTI is given in Table 1. The failure temperatures for the ANL tests were used to recalculate the unaged flow stress curve of the electrosleeve material using the Hall-Petch model (with n=0.33) and the effective m_p factors from Fig. 2b and Eq. 2. The revised unaged flow stress curve is compared with the previously estimated flow stress curve (Fig. 4) in Fig. 8. Note that the revised curve has a different shape and falls below the earlier estimated curve.

An examination of Table 1 shows that the geometries of the Electrosleeved tubes have some variations. An upper bound to the predicted failure temperatures was obtained by using the following:

Tube thickness = 0.051 in., sleeve thickness = 0.040 in. and notch depth=0.048 in.,

and a lower bound was obtained by using the following:

Tube thickness =0.049 in., sleeve thickness = 0.035 in. and notch depth=0.049 in.

The two bounds together with the test data are plotted in Fig. 9a. In cases where the notch depth was less than the full thickness of the parent tube wall, an effective flow stress for the ligament (average flow stress weighted by thickness) was used. Both the test data and the model indicate that the decrease in failure temperature with notch length saturates at a notch length of ≈ 3 in. and no significant additional decrease of failure temperature should occur at longer notch lengths. The tube-to tube variations in geometry give rise to a significant difference between the two bounds, and a much better correlation between the predicted and the observed failure temperatures is obtained if the actual geometry for each specimen is used in calculating the predicted failure temperatures (Fig. 9b).

5.3 Revised failure predictions for the FTI tests

Fig. 10 shows a comparison of the failure temperatures as reported by FTI and the two bounds based on the same bounding geometrical assumptions as in Fig. 9a. All the tests tend to fall near the lower bound curve (concave upwards), which is not surprising because the thickness of the electrosleeve was close to the lower bound thickness assumed for the curve. The predicted failure temperatures (using actual geometry and actual temperature ramp) are within 15°C of the observed values (Fig. 9b). The almost linear variation of the experimental failure temperatures of the Electrosleeved tubes with notch length (the curve is actually slightly concave downwards) in Fig. 10 is fortuitous because of the way the geometry and temperature ramps varied in these tests. Both the model predictions and ANL data (Fig. 9a) clearly show that the deleterious effect of notch length should level off beyond a notch length of 3 in.

6. Predicted Failure Temperatures for Severe Accidents

6.1 Case 6 (SBO severe accident)

As mentioned earlier, calculations were performed for the reference 7/8 in. dia. tube (wall thickness = 0.050 in. and electrosleeve thickness = 0.038 in.) with 100% throughwall cracks of various lengths in the parent tube subjected to the temperature and pressure histories that closely simulate Case 6 of Ref. 2. The results, plotted in Fig. 11, show that failure temperatures for 100% throughwall 3, 2, 1 and 0.5 in. long cracks are 641, 647, 691 and 792°C, respectively. Since the surge line failure occurs when the tube temperature = 684°C, cracks ≤ 1 in. long are predicted to survive the Case 6 transient. Note that the reduction is 6°C in failure temperature going from a crack length of 2 in. to 3 in. and an additional 6°C from 3 in. to 6 in. Thus although the ANL correlation suggests that the failure temperature continuously decreases with increasing crack length, from a practical view point the additional decrease beyond a crack length of 3 in. is negligible. Ii is anticipated that if the cracks were part-throughwall rather than 100% throughwall, the predicted failure temperatures would be significantly higher than those indicated in Fig. 11. Such calculations are currently in progress.

6.2 Case 20C (SBO with pump seal leakage)

The variation of temperature and pressure differential during an SBO severe accident with pump seal leakage (Case 20C) is shown in Fig. 12a. The variations of temperature, ligament flow stress and ligament stresses for various crack lengths with time are plotted in Fig. 12b for the reference 7/8 in. dia. tube (wall thickness = 0.050 in. and electrosleeve thickness = 0.038 in.) with 100% throughwall cracks. Since none of the ligament stresses exceed the flow stress before surge line failure, tubes with 0.5, 1 and 2 in. cracks are predicted to survive this transient.

7. Discussions and Conclusions

A flow stress-based model has been developed for predicting failure under expected severe accident transients. The model accounts for the loss of flow stress of the electrosleeve with aging at high temperatures. Aging has been simulated using a grain growth model and the hardness data supplied by FTI on electrosleeve material aged at high temperatures. Thus, there is reason to expect some uncertainty in the calculated loss of flow stress with aging. However, analyses showed that any inaccuracy in the model for the nucleation of loss of flow stress should not influence the predicted failure temperatures for the severe accident scenarios considered in this paper. FTI has suggested that the flow stress of Ni-200 at high temperature should provide a reasonable estimate for the flow stress of electrosleeve after grain growth. A comparison of flow stress data of Ni-200 and Ni-201 with the calculated flow stress of the electrosleeve for the Case 6 ramp rate (including the effect of aging) is shown in Fig. 13. The data for Ni-201 extends only to 649°C. The two FTI data points at 593°C and 760°C on aged electrosleeve material fall quite close to the Ni-200 curve. In the temperature range of interest for severe accidents, i.e., >650°C, the calculated aged flow stress curve is close to but a little below the Ni-200 flow stress curve. Note that the FTI data at 760°C on aged electrosleeve falls below the Ni 200 curve and is closer to the calculated flow stress curve. Thus, the present estimates for loss of flow stress with aging are consistent with the FTI assumption for the severe accident transient.

Finite element analyses were conducted to validate the m_p factor used in the model for calculating average ligament stress in single layer shells with part-through axial cracks. The same model showed that the m_p factor for the electrosleeve ligament in a 100% throughwall axial crack is reduced when the flow stress of the electrosleeve is reduced compared to that of the parent tube. The reduction is greater for shorter cracks. Therefore, a flow stress and crack length-dependent correction factor was applied to the m_p factor calculated with the ANL correlation that was developed originally for single layer shells.

Eleven high temperature tests simulating an SBO (Case 6) pressure and temperature ramp have been conducted on notched Electrosleeved tubes supplied by FTI. The test results indicate a leveling off of failure temperature with crack length beyond 2 - 3 in., which is consistent with the FEA results. The flow stress data at low temperatures supplied by FTI together with the ANL test results were used to derive an unaged flow stress curve of the electrosleeve from room temperature to high temperatures. The unaged flow stress curve was used in the model for predicting failure. All the test data fall within the upper and lower bounds calculated on the basis of limiting geometrical parameters observed in the specimens. Also, high temperature test data on notched unsleeved as well as notched Electrosleeved tubes reported by FTI can be predicted reasonably well by the flow stress model.

The reference Electrosleeved tube with throughwall axial cracks ≤ 1 in. in the parent tube is predicted to survive the postulated SBO (Case 6) transient until surge line failure. The same tube with throughwall axial cracks of any length ≤ 3 in. is predicted to survive the severe accident transient Case 20C until surge line failure. These maximum survivable crack lengths should be increased if the crack depth is less than 100% throughwall.

The failure temperature of the Electrosleeved tube under any severe accident scenario can be increased by increasing the electrosleeve thickness. For example, the effect of an increase of electrosleeve thickness from 0.038 in. to 0.043 in. on the ligament failure temperature of tubes with various throughwall axial cracks subjected to the reference Case 6 SBO ramp is shown in Fig. 14. There is an increase of 20 - 30°C in the failure temperature relative to that of the reference Electrosleeved tube depending on the crack length.

The proposed model with the unaged flow stress curve of the electrosleeve material reported here are valid for temperature ramps that are not significantly different from the ramp rate (12.5°C/min) used in the ANL tests because creep effects are neglected in the model. The rate effect that is predicted by the model is due to grain growth only. Predicted failure temperatures at ramp rates significantly different from 12.5°C/min will be accurate if grain growth effects predominate creep effects.

Acknowledgement

Helpful discussions with Drs D. R. Diercks and W. J. Shack of Argonne National Laboratory and Dr. J. Muscara of the USNRC are acknowledged. Thanks are also due to Framatome Technologies, Inc. for providing the Electrosleeved specimens and Mr. J. Franklin of Argonne National Laboratory for conducting the high temperature tests.

References

1. S. Majumdar, W. J. Shack, D. R. Diercks, K. Mruk, J. Franklin, and L. Knoblich, "Failure Behavior of Internally Pressurized Flawed and Unflawed Steam Generator Tubing at High temperatures – Experiments and Comparison with Model Predictions," NUREG/CR-6575, Argonne National Laboratory, Argonne IL 60439, 1998.

2. SGTR Severe Accident Working Group, "Risk Assessment of Severe Accident-Induced Steam Generator Tube Rupture," NUREG-1570, U.S. Nuclear Regulatory Commission, Washington DC, 1998.

3. S. Taira, "Lifetime of Structures Subjected to Varying Load and Temperature," in *Creep in Structures*, N. J. Hoff, Ed., Springer-Verlag, Berlin, pp. 96-119, 1962.

4. E. O. Hall, Proc. Phys. Soc. (London), Vol. 64B, p. 747, 1951.

5. N. J. Petch, J. Iron Steel Inst. (London), Vol. 173, p. 25, 1953.

Table 1 Summary of simulated severe accident tests conducted at ANL and FTI on notched Electrosleeved tubes

Test No.	Notch length (in.)	Notch depth (in.)	Tube wall thickness (in.)	Electrosleeve thickness (in.)	Failure temperature (°C)
BTF-21	0.5	0.0490	0.0490	0.0410	807
BTF-13	0.5	0.0492	0.0510	0.0400	806
BTF-4	1.0	0.0482	0.0510	0.0390	722
BTF-10	1.0	0.0500	0.0520	0.0380	724
BTF-14	1.0	0.0490	0.0500	0.0390	714
BTF-19	2.0	0.0490	0.0510	0.0400	680
BTF-22	2.0	0.0495	0.0500	0.0380	653
BTF-20	2.0	0.0490	0.0510	0.0370	653
BTF-18	3.0	0.0503	0.0505	0.0395	643
BTF-17	3.0	0.0493	0.0510	0.0350	630
BTF-5*	3.0	0.0490	0.0490	0.0440	673
BTF-23**	0.5	0.049	0.050	0.035	731
BTF-25**	1.0	0.051	0.051	0.036	691
R.5.2**	2.0	0.051	0.050	0.036	611

*One tip of the notch in this specimen was about 0.1 in. from the end of the electrosleeve

**these tests were conducted by FTI

Table 2 Observed and initial predictions of failure temperatures for the FTI high temperature tests on pressurized unsleeved and Electrosleeved tubes.

	Electrosleeved Alloy 600 tube			Unsleeved Alloy 600 tube		
	0.5 in. 100%TW	1.0 in. 100%TW	2.0 in. 100%TW	2.0 in. 50% TW	50% uniform thinning	Un-degraded
Observed failure temp.	731°C	691°C	611°C	727°C	724°C	82 min*
Predicted failure temp.	766°C	728°C	682°C	738°C	726°C	164 min**

* this test was held at 764°C until it failed after 82 min

** predicted by the creep rupture model of NUREG/CR-6575

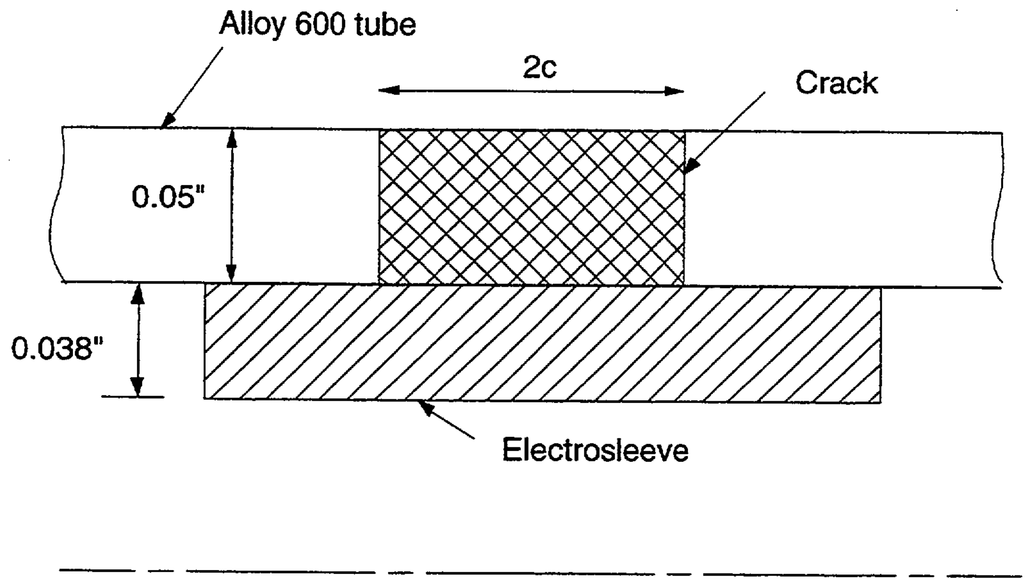

Fig. 1. *Reference geometry for Electrosleeved steam generator tube with an axial crack.*

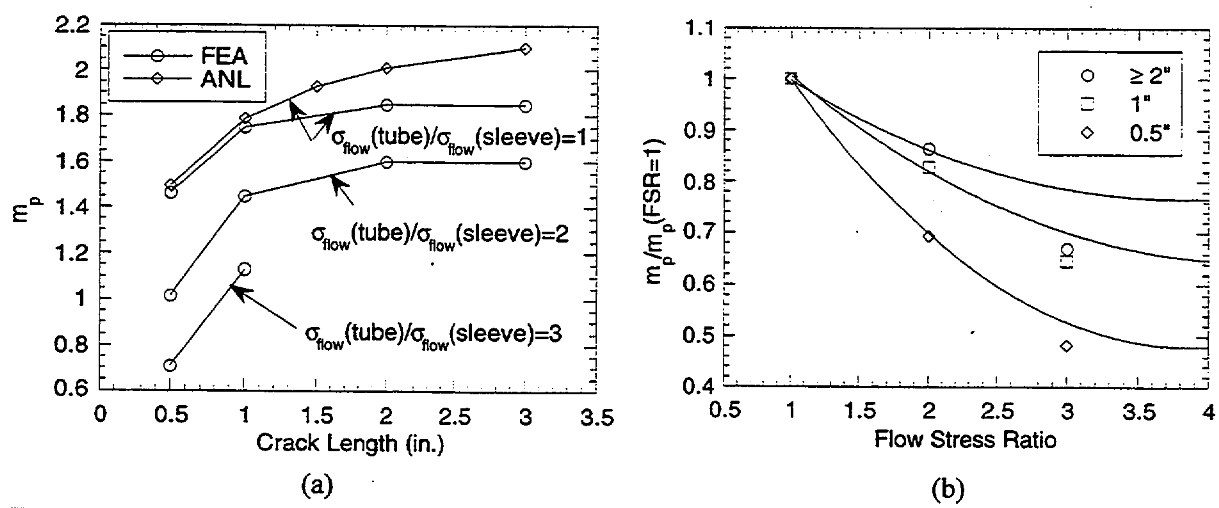

Fig. 2. *(a) Comparison of m_p values calculated by ANL correlation with those by FEA for ratios of flow stress of Alloy 600 and the electrosleeve of 1, 2, and 3, and (b) Assumed variation of the m_p reduction factor with flow stress ratio (FSR). Symbols denote values calculated by FEA.*

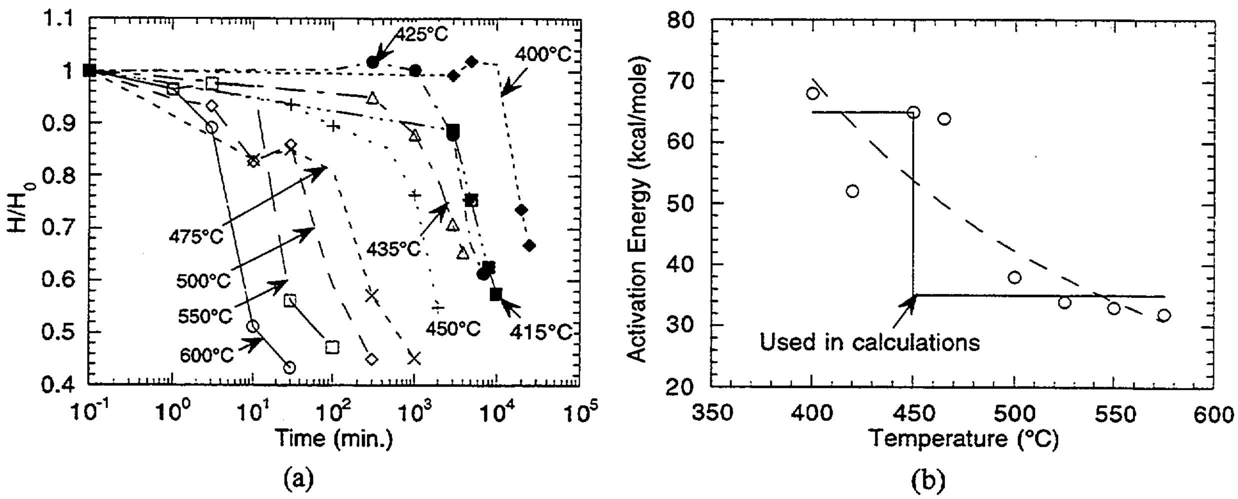

Fig. 3. *(a) Variations of normalized Vickers Hardness Number (VHN) of the electrosleeve material with time under isothermal aging at various temperatures and (b) variation of activation energy for the reciprocal of the time to onset of rapid reduction of flow stress (or grain growth) with temperature.*

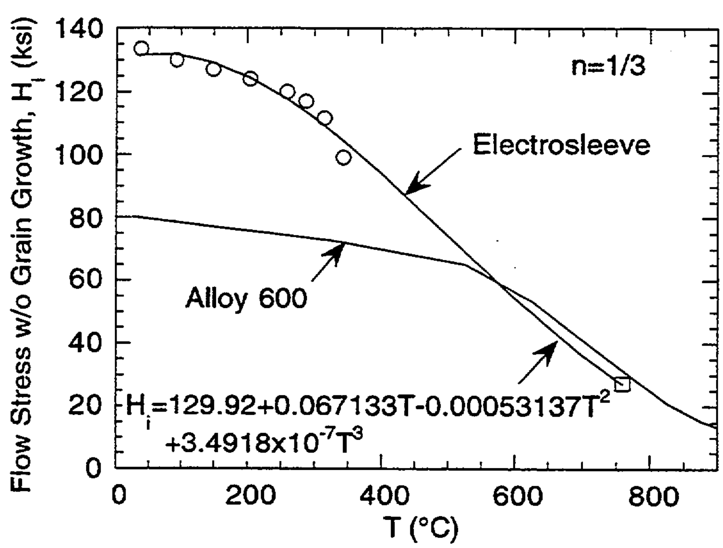

Fig. 4. *Flow stress (without aging) vs. temperature plot for electrosleeve material and Alloy 600. The electrosleeve data (square symbol) at 760°C was estimated from tensile data on a single specimen pre-aged and tested at 760°C, using n=0.33. Note that this flow stress curve of the electrosleeve was subsequently modified on the basis of ANL tests.*

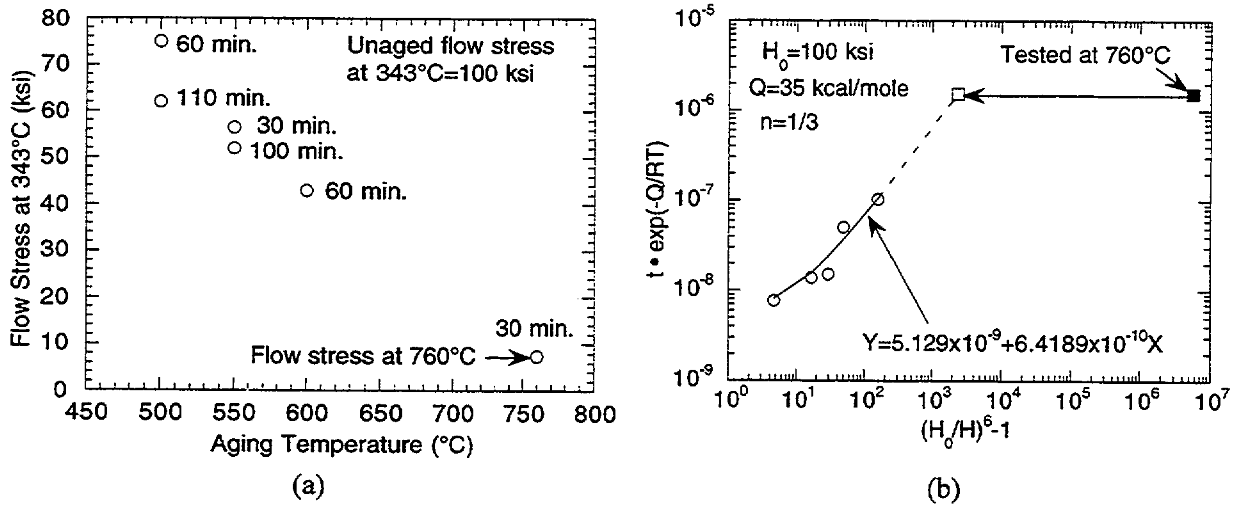

Fig. 5. *(a) Flow stress data on the electrosleeve material pre-aged for various times at high temperatures. All the tensile tests were conducted at 343°C, except for the test on the specimen pre-aged at 760°C, which was conducted at 760°C and (b) normalized flow stress vs. time-temperature parameter plot for electrosleeve material.*

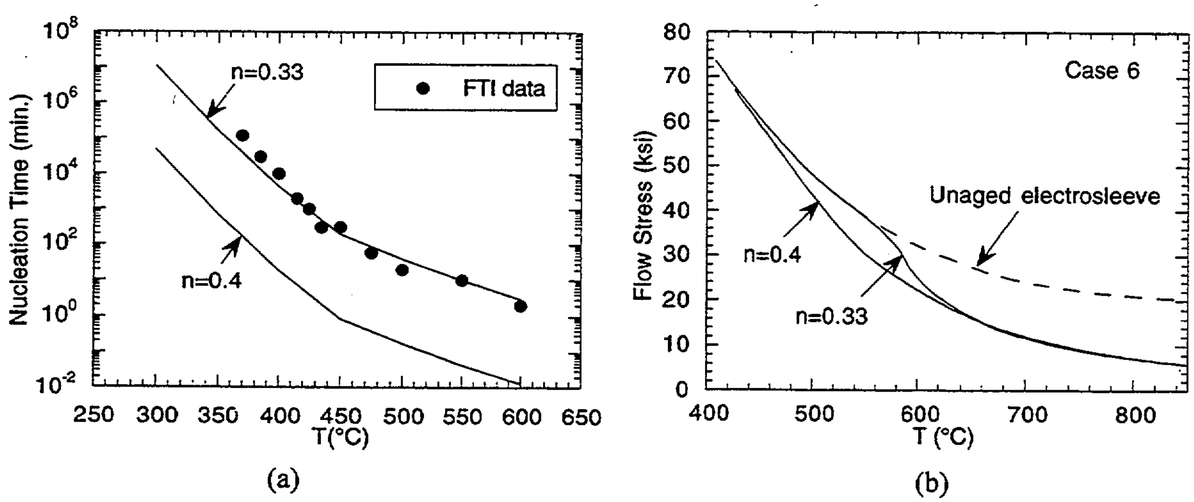

Fig. 6. *(a) Variation of calculated "nucleation" times to onset of rapid loss of flow stress (or grain growth) under isothermal aging with aging temperature for Hall-Petch exponents of n= 0.33 and n = 0.40, using a temperature-dependent activation energy given by the step function in Fig. 3b. Also shown are nucleation times for rapid loss of flow stress derived from the FTI data shown in Fig. 3a, and (b) effect of exponent n on the loss of flow stress of the electrosleeve subjected to Case 6 temperature transient.*

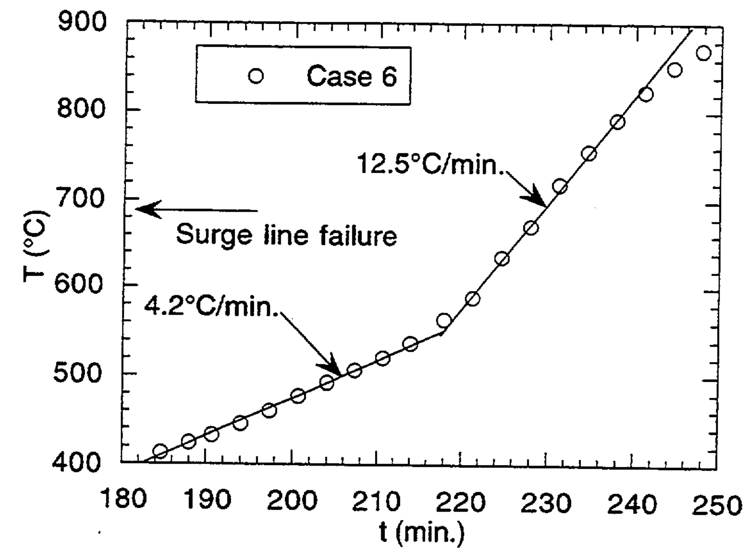

Fig.7. Calculated variation and ANL test simulation of temperature during an SBO (case 6) severe accident transient.

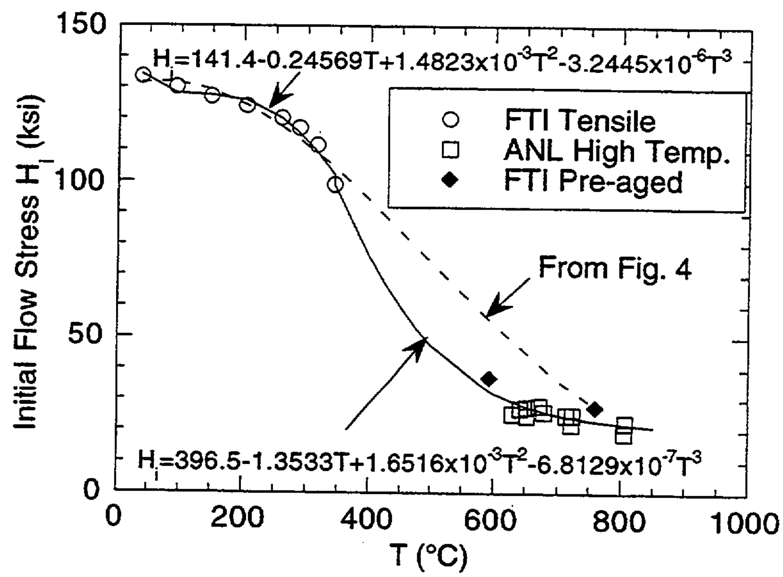

Fig. 8. *Original unaged flow stress curve (dashed line) of the electrosleeve estimated from FTI tensile data before the ANL tests were conducted and revised unaged flow stress curve (solid line) of the electrosleeve calculated using the ANL tests.*

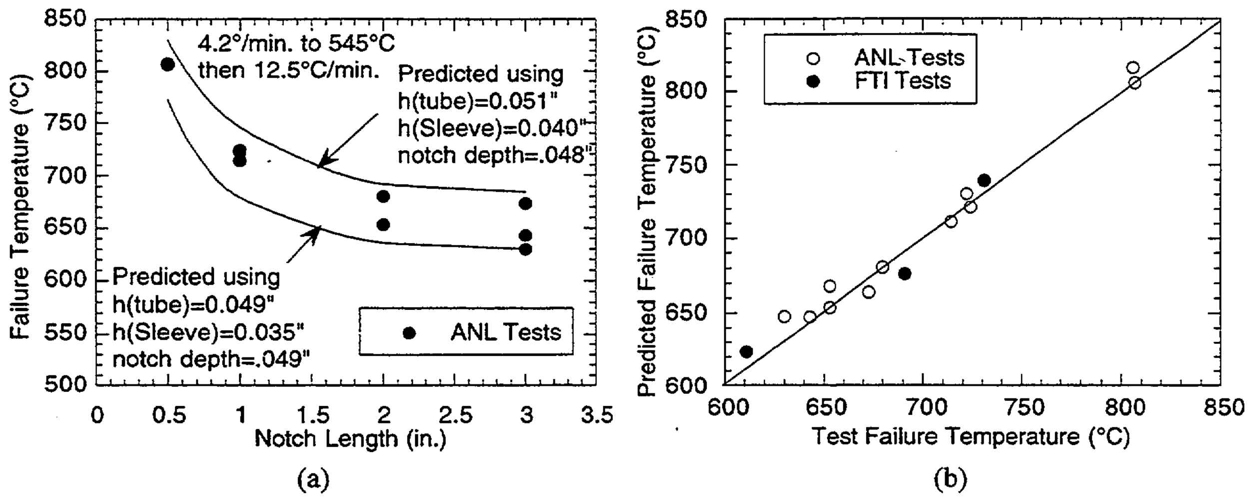

Fig. 9. *(a) Variation of ANL test failure temperatures and predicted upper and lower bounds to the failure temperatures with notch length and (b) observed vs. predicted failure temperatures of the ANL and FTI tests using actual notch and electrosleeve geometry and, for the FTI tests, the actual test temperature ramp.*

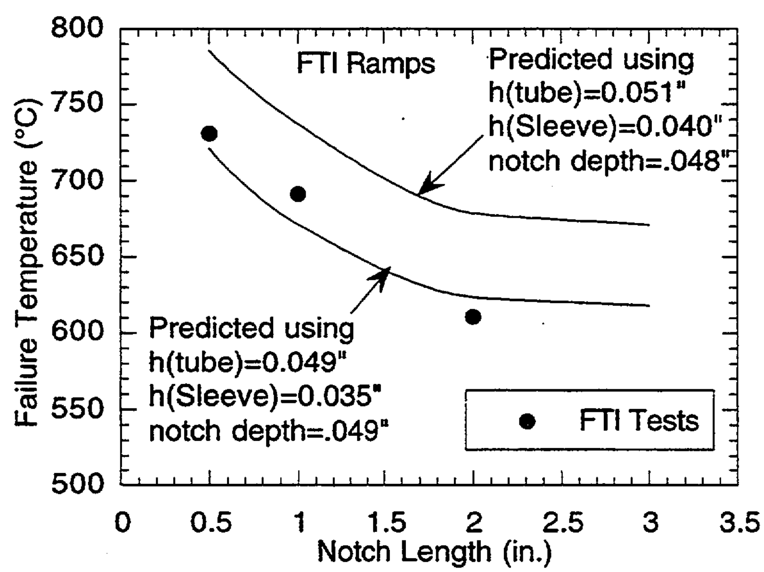

Fig. 10. *Variation of FTI test failure temperatures and predicted upper and lower bounds to the failure temperatures with notch length.*

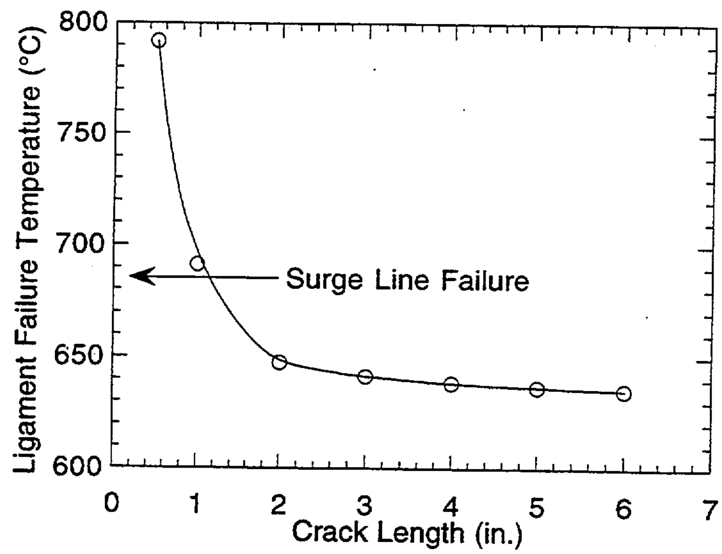

Fig. 11. *Predicted ligament failure temperature during the SBO severe accident (Case 6) vs. crack length for 100% throughwall cracks in the parent tube. Also shown is the tube temperature at the time of surge line failure.*

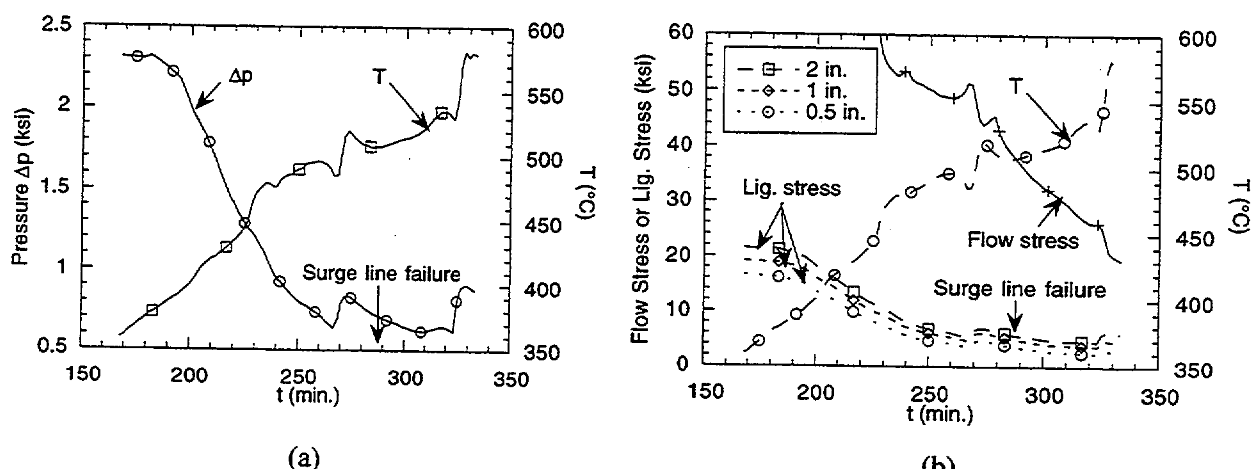

Fig. 12. *(a) Variation of temperature and pressure during SBO with pump seal leakage (case 20C) severe accident transient. Also shown is the time at surge line failure and (b) time variations of temperature, flow stress of the electrosleeve, and average ligament stresses predicted for 100% throughwall cracks of length 0.5 in., 1 in., and 2 in. in the parent tube during the severe accident Case 20C. The flow stresses for times less than ≈228 minutes are > 60 ksi. Note that the ligament stresses are well below the flow stress up to surge line failure.*

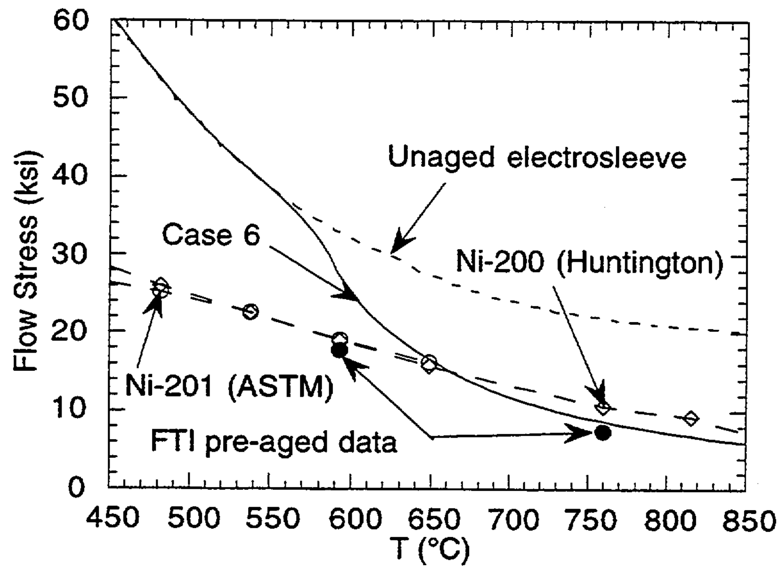

Fig. 13. Comparison of calculated flow stresses (including aging) of electrosleeve (solid line) with flow stress data (long dash line with open symbols) of Ni 200 (Huntington) and Ni-201 (ASTM). Also shown are the flow stress of the unaged electrosleeve (short dashed line) and two FTI flow stress data (filled circle) on 30-min aged specimens.

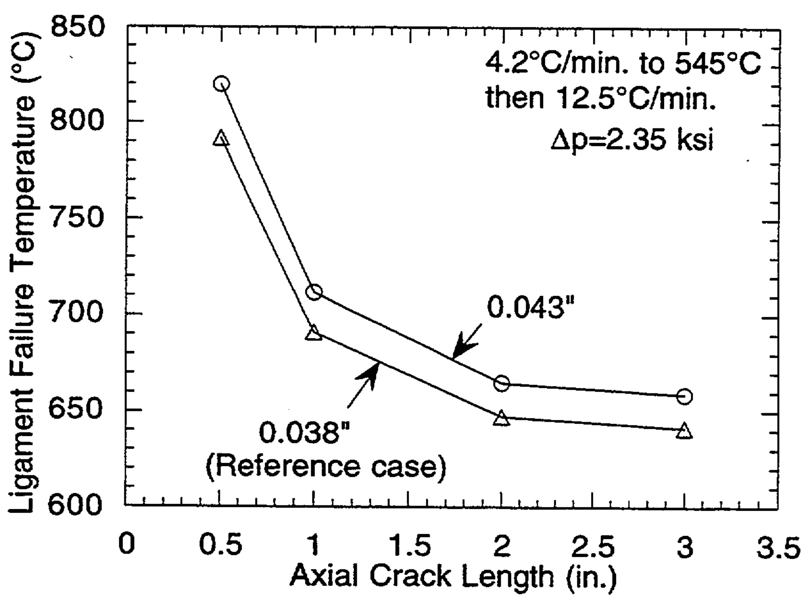

Fig. 14. Effect of electrosleeve thickness on the predicted ligament failure temperature of a tube with throughwall axial cracks.

RISK-INFORMED CONSIDERATIONS IN THE EVALUATION OF STEAM GENERATOR TUBE INTEGRITY

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In 1995, the NRC issued its PRA Policy Statement, which stated that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art. Subsequently, the agency issued Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," July, 1998. The process for applying risk in regulatory matters is outlined in Reg Guide 1.174 and a number of companion guides covering specific applications. Reg Guide 1.174 also defines general safety principles and a set of guidelines which can be used for changes to the design basis of plants.

At first, the main use of this guidance was for the review of plant-specific license amendments, such as changes to technical specifications, and for risk-informing broad regulatory programs, such as in-service inspection (ISI) and in-service testing (IST). As the staff gained experience with the risk-informed approach, it was applied in wider arenas, such as the new oversight process for plant performance assessment. There is also a large effort in place to consider significant changes to the NRC's reactor regulations based on risk-informed thinking.

In recent years, the nuclear industry has experienced various forms of steam generator tube degradation which have adversely affected operational performance. This issue has led to the development of improved techniques for detection and repair of tube cracking, as well as new methods for analyzing operational performance and predicting the extent of degradation for tubes left in service over the next operating cycle. In accordance with NRC requirements, licensees perform these assessments in the context of normal operations and design basis accident conditions, and do not explicitly consider severe accident risk. Nevertheless, it is well understood that steam generator tubes play an important role in certain classes of severe accidents, and are an important contributor to controlling overall plant risk. Consequently, the NRC has considered severe accidents in its evaluations of generic industry initiatives and plant-specific applications related to tube cracking.

This paper describes how severe accident technical issues have been addressed in the NRC's evaluation of the risk implications of steam generator performance. In addition, the paper discusses an ongoing agency initiative to define the regulatory framework in which risk considerations can be applied to steam generator degradation and other licensing issues.

New Perspectives on Steam Generator Tube Behavior:

The original regulatory framework for oversight of steam generator tubes was predicated on the view that Inconel tubes would degrade by a gradual process of thinning. The methods of in-service inspection, the technical specifications and the criteria for tube repair were all based on this assumption. The repair criterion was set at 40% through-wall degradation.

More recently, the industry became aware that cracking is the more dominant mode of tube degradation. This has raised questions about the method and frequency of inspection, the criteria for repair and the best framework for regulatory oversight. Cracking is more difficult to detect and size than uniform thinning. Moreover, the depth of cracks can grow much faster than the rate of wall thickness loss due to thinning.

Risk-Informed Regulation:

Regulatory Guide 1.174 states five safety principles that govern the use of PRA in reviewing proposed changes to the design basis:

- The proposed change must meet the current regulation, unless it is explicitly related to a requested exemption,
- The proposed change must be consistent with the defense-in-depth philosophy,
- The proposed change must maintain sufficient safety margins,
- When a proposed change results in an increase in core damage frequency or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement, and
- The impact of the proposed change should be monitored using performance measurement strategies.

Steam generator tube degradation has important implications for risk. For some accident scenarios, the tubes represent two of the three radiation barriers, the RCS and containment. Steam generator tube ruptures have occurred during normal operation on several occasions over the past twenty years, and have resulted in release of RCS coolant activity to the environment. PRA studies have shown that such incidents have the potential for leading to core damage with bypass of the containment boundary. Core damage accidents of that category are classified as large early releases, and are predicted to result in significant offsite consequences.

An additional risk concern is the potential for tube failure during core damage accidents initiated by other causes. A class of core damage accidents called "high-dry" sequences can produce thermal-hydraulic conditions that would fail degraded tubes. This class of accidents includes station blackout scenarios. The failure of the tubes would result in bypass of containment at the time when the core was about to melt.

Past risk assessments have concluded that the frequency of core damage accidents initiated by steam generator tube rupture is acceptably low for PWRs operated in the U.S. Moreover, severe accident studies have concluded that the failure of tubes as a result of "high-dry" core melt sequences is unlikely without substantial degradation of the tubes.

Risk Implications of Tube Cracking:

The emergence of tube cracking as a primary mode of tube degradation presented two potential problems for plant risk. The first was the possibility that the frequency of spontaneous steam generator tube ruptures might increase, thereby raising the probability of a core melt accident. This frequency can be monitored over time, and any adverse trends can be addressed if they emerge. It is worth noting that the frequency of tube ruptures has decreased in recent years, although the trend may not be statistically significant. One factor in this decrease may be a better understanding of tube flaw behavior, and a more realistic method of monitoring tube degradation. That remains to be seen. In any case, rupture during normal operation has not been the main focus of NRC concern.

The potential for tube rupture induced by design basis differential pressure transients and severe accident thermal transients has been the agency's more recent concern. The more difficult analytical issue is that the probability of induced tube ruptures during high-dry core melt sequences might be higher than previously believed. Existing severe accident thermal-hydraulic calculations could no longer be counted on to assure a margin of safety for the tubes.

As the NRC and the industry set out to define a new regulatory framework for oversight of tube integrity, they did so in the context of design-basis accidents and challenges. This is the normal method for NRC to control the design of nuclear plants. Risk assessments are performed based on the plant as found, assuming it meets the design basis criteria. In this case, the NRC staff sought to take risk insights into account in defining the design basis requirements. Through this approach, the guidance in NEI-97-06 was based on design basis accidents, but it was derived in a manner that controls risk increases above levels permitted by current tube integrity requirements.

Proposed Regulatory Treatment of Tube Integrity:

NEI 97-06 embodies a performance based approach to assuring tube integrity. It defines performance criteria for the limiting tube at the end of an operating cycle. The structural criterion requires that the limiting tube be capable of withstanding three times the normal operating pressure and 1.4 times the pressure experienced in a steam line break design basis accident. The leakage criterion is that the sum of the leakage experienced in a design basis accident (excepting the tube rupture accident) should be less than or equal to 1.0 gallon per minute.

The document calls for a program to periodically monitor performance against these criteria at each refueling outage. The inspection interval is to be adjusted to achieve the performance criteria. While NEI 97-06 does not specify an analysis method for adjusting the inspection interval, current practice is for licensees to perform an operational assessment using probabilistic fracture mechanics methods. Ultimately, however, it is the results of the performance monitoring program that will govern long term risk.

Tubes with flaws in the free span are to be repaired on detection, unless the flaws can be sized and the growth rates can be predicted to meet the performance criteria at end of cycle. NEI 97-06 allows for alternative repair criteria to be used with approval from the NRC staff. This approval may be plant-specific or generic, and will include a risk-assessment, when warranted.

Currently, two methods of repair for free-span cracks are generically acceptable to the NRC: plugging and Inconel sleeving. Electrosleeving has been approved on an interim basis for the Callaway plant. The NEI guidance specifies that alternate repair methods may be used if the NRC has approved them generically or on a plant-specific basis. Plants may not use a repair method that has received plant-specific approval for another plant.

Risk Perspective on NEI 97-06:

The staff has concluded that the provisions of NEI 97-06 provide an adequate means to control the severe accident risk associated with steam generator tube integrity within the principles and guidelines of Reg Guide 1.174. Controlling the probability of tube failure in a severe accident preserves the defense-in-depth philosophy by ensuring a low likelihood of multiple barrier failures. Conformance with the performance criteria provide sufficient margin to control the frequency of steam generator tube ruptures which might lead to core damage. Furthermore, the licensees' programs to maintain the strength margin provided by the performance criteria, together with the low frequency of "high/dry" type severe accidents, appears to adequately limit risk from induced tube failures in the event that the core is damaged by some other cause.

The performance criteria and associated licensee monitoring program meet the fifth principle of Reg Guide 1.174; namely, using performance measurement strategies to monitor the impact of proposed changes.

In sum, this new approach to steam generator integrity preserves defense-in-depth and sufficient safety margins, maintains core damage frequency and risk within acceptable levels and provides a robust method for monitoring performance.

Outline of a RCCV Seismic Proving Test Results

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Abstract

Concrete containment vessels in nuclear power plants are the final barriers to secure radioactive materials from releasing to surroundings. CCVs are required to maintain airtight even under severe earthquakes. In the past, several static and/or dynamic tests were conducted to confirm the structural reliabilities of RCCVs under severe earthquakes. But in those tests, air-tightness, was not verified dynamically under seismic test.

Nuclear Power Engineering Corporation (NUPEC) conducted the shaking table tests of RCCV using large-scale (1/8) model with liner plate. The configuration scale is 1/8, and wall thickness is 1/10 scale considering to fail the specimen and the construction capabilities. The liner plate thickness is 1/4 scale determined by welding capabilities. The test was carried out employing the large-scale and high-performance shaking table at Tadotsu Engineering Laboratory of NUPEC in 1998 , 1999.

The shaking test of RCCV was completed this April and test data were evaluated. Evaluation of actual plant of RCCV based on the test results is now under the way.

1. INTRODUCTION

The objective of the test is to prove structural integrity as well as functional soundness of RCCV. The structural integrity up to the extreme design earthquake S2 was confirmed by investigating shear stress of concrete, strain of rebar and strain of liner. The functional soundness was also confirmed by measuring air-tightness up to S2. The test model finally collapsed with the shear failure mode by large seismic excitation and the ultimate strength of RCCV was obtained.

2. TEST RESULTS

2.1 Outline of the Test

The RCCV portion of an ABWR prototype building is separated from the building to make the test model. The form of the upper part of the RCCV model is simplified. This modeling is necessary to enable the final collapsing stage tests within the shaking table's capacity. A part of the floor slab is also modeled to simulate the restraining effect of the outer building on the response to the earthquake load and the inner pressure. Masses are set at the top part of the model to make the acceleration and the stress the same as that of the prototype. Simulations confirmed the earthquake response consistency between the model and the prototype.(refer Fig.1, Fig.2 and Table 1)

Input wave forms to the shaking table and the test model are artificial earthquake motions based on the MITI report of the Second Modified Standardization Survey in Japan. Figure3 shows wave forms and response spectra of the extreme S2 design earthquake used for the proving test. The time scale is set $1/\sqrt{8}$ in accordance with the similarity law. For the safety margin tests, input wave was adopted horizontal only and the amplitudes were increased in integer multiples of S2. Measurements of 300 chs of acceleration, stress and displacement data were acquired.

The transition of the 1st peak frequencies and damping ratios up to S2 are shown in Fig.4 and Fig.5 . At first, the decrease in frequency is remarkable, but afterward it is not. The damping ratios show some differences depending on the methods used. At first they are about 1% and then they increase to 5 to 6% with small variation.

After the 5S2 test, 9S2 excitation was performed and specimen failed with shear mode.

2.2 Maximum response stress and strain

Relationship between maximum value of base shear and deformation angle of the specimen is shown in Fig.6. Structural behavior changed from elastic range to elasto-plastic range at S1 excitation.

The average shear stresses were 32.3kgf/cm2 for 1.1S1(H+V) and 52.0kgf/cm2 for 1.1S2(H+V). These values are less than the design strength of 64.0kgf/cm2. During 1.1S2(H+V) excitation, the largest rebar strain was 2143 millionths, which was observed in an outer vertical rebar at the bottom of the 90 degree side of the shell wall. This value is less than the yield strain. The strain in the horizontal rebars and the rebar around the L/D access tunnel openings were also less than the yield value. The strains of liner were within a yield strain but some area of the liner plate around the openings yielded. After detailed investigation of these areas, it was found that yielding occurred at discontinuous portion of the thickness, so the integrity of the liner should be evaluated by averaged peak values. An evaluation of the liner plate using averaged peak values verifies that the liner plate maintains its integrity.

Vertical rebars at the base portion of wall were almost of all yielded at 5S2 but horizontal rebars were within the yielding strain.

2.3 Concrete cracks

Concrete cracks on the shell wall surface observed after 1.1S2(H+V) excitation are shown in Fig.7. These cracks were observed during the leak test, when the internal pressure was A.P. plus 2.15kgf/cm2. As the model suffered such internal pressures in the pressure proving test and leak tests, tangential ring tension cracks caused were observed over the whole shell. Bending cracks occurred at the shell flange(i.e. on the 90 degree and 270 degree sides). Diagonal shear cracks occurred around the shell web(i.e. around the 0 degree and 180 degree sides). Shear failure occurred at the side of the openings and compressive failure occurred at the base portion of wall- flange same time.

2.4 Changes in Stiffness Deterioration Rate

Stiffness deterioration rates were evaluated by taking the ratio of the secant modulus of elasticity to the initial value. The secant modulus of elasticity to the initial value. The secant modulus of elasticity was evaluated at the maximum and minimum points of both force and deformation. The changes in stiffness deterioration rates are shown in Fig.8. This shows that the horizontal stiffness decreased to 85% of its initial value after the internal pressure test, to 50% after 1.3S1(H) excitation and remained at about 30% after LOCA+1.2S2(H+V) excitation. And after the failure test stiffness decreased to less than 20% of its initial value.

2.5 Pressure Test Results

The pressure test was carried out at 3.07kgf/cm2. No abnormal deformations were observed in the test results. Cracks were mainly distributed at the middle of shell, and measured crack width was under 0.05mm. The maximum reinforcements strain was observed at the upper side of the opening, where the first crack was observed, and the strain was 469 millionths.

2.6 Leak Test Results

Leak test were performed after a pressure test and after seismic tests. The internal pressure was held at about 3 kgf/cm2 for 1st leak test and about 2.4kgf/cm2 for after 2nd leak test. The change in leakage quantity is shown in Fig.9. The internal pressure and leakage quantity showed constant values. Each leak tests carried out during the seismic vibration tests before and after showed the same leakage rate. After the collapse, the local leak rate test was performed on the welding line of the liner plates and the largely distorted area adjacent to the openings. No leakage was confirmed at liner plates. This leak rate test demonstrated that there was no leakage unless test model is collapsed. So the reliability of the leakage boundary in the test model was confirmed through this seismic proving test of RCCV.

3. SIMULATION ANALYSES

3.1 Analytical Model

Non-linear static analyses for the pressure test,S1(H) and S2(H) verification tests were carried out by a 3-dimensional FEM model, for comparison with test results. The shape of the FEM model is shown in Fig.10 . Six layered shell elements were used for the concrete part. Liner plates inside the vessel were modeled by shell elements. The base of the model was assumed to be fixed to the ground. Table2 summarizes the material properties utilized in the modeling. And the time history nonlinear response analysis were carried out by lumped mass model. Analytical results of dynamic analyses showed good agreement with the test.

3.2 Pressure Test Analysis

The many cracks observed in this test suggested the existence of initial tensile stress due to concrete shrinkage Fig.11. Pressure test was carried out at the beginning of the proving test sequence. Rebar strains measured with wire strain gauges show that an additional stress of 8.5kgf/cm2 caused these cracks. For simplicity, the tensile strength of concrete is supposed to be 8.5kgf/cm2 in this analysis. The analytical results are compared with the test results shown in Fig.12(1) and Fig.12(2), of radial displacement of the 2nd story wall. The average of two mid-height points at 90deg. And 270deg. of longitude is shown in Fig.12(1). Fig.12(2) shows the average of the points at the top of two openings located at 0deg. and 180deg. of longitude. The analysis results correspond closely to the test results.

3.3 Analysis of S1 and S2 Verification Test

The analytical model is the same as that for the pressure test simulation, except for the concrete properties. It is found that the overall horizontal stiffness of the specimen declined to about 85% of its initial value after the pressure test. Therefore, Young's modulus for the concrete is reduced to 85% of the elastic value. A concrete tensile strength of, ft=17.1kgf/cm2 obtained from the material test, is used. It is assumed that the initial tensile stress of concrete is reduced to zero through cracking in the pressure test. Fig.13 compares the hysteresis loops of the 1.3S1(H) and 1.1S2(H) excitation tests with the static analysis results. It seems that the static analysis curves agree with the envelope curves of the dynamic test results.

3.4 Dynamic Analyses

Fig.14 shows the dynamic analysis model for horizontal direction and Table 3 shows the simulation analysis cases. The initial stiffness and damping values and frequency transfer functions were compared to those from the test results before starting each time history nonlinear analysis. Frequency transfer functions and those obtained from the test results are compared as shown in Table 4. The peak frequencies matched with those of the test results.

Time histories of horizontal acceleration at the upper slab obtained by the analysis are compared with the test results in Table 5. The hysteresis loops between forces and deformations obtained by the analyses were compared with the test results in Table 6. The time histories of the horizontal acceleration shows that the amplitudes from the analyses during the excitation correspond well with those from the test results. Hysteresis loops show that the peak values of shear force from the analysis are slightly larger than the test results for S1 level excitations, but this relations are opposite for S2 level excitations. However, the maximum deformations from the analysis are slightly smaller than the test results for all excitations.

4. SEISMIC MARGIN TEST RESULTS

Seismic margin tests for RCCV were carried out after S2 design level earthquake excitation. Acceleration of the input wave was increased sequential 2S2, 3S2, 4S2 and 5S2 after S2 excitation. Vertical rebar strains at the base portion of the wall were almost of all yielded at 5S2. After 5S2 test 9S2 test was carried out, shear failure at the side of opening and compressive failure at the base portion of the flange wall part occurred. The safety margin of S2 to the collapse will be in terms of shear force, input energy and so on. From the observation result of liner plate and concrete after failure, leak rate test results after each seismic margin test and the local leak rate test after failure, It was found that the RCCV has sufficient structural and functional safety margins.

5. CONCLUSION

A series of seismic proving test of an RCCV was carried out. As for the test results up to the S2 design level. The results confirmed the seismic safety of the RCCV based on the shear stress in the concrete shell wall, rebar strains, and liner plate strains. Further, many test results were obtained that are useful for the evaluation of RCCV integrity.

A pressure test was conducted prior to the seismic tests. No excessive concrete cracking was observed, rebar strain was sufficiently smaller than yielding strain and measured displacement showed good agreement with nonlinear static analysis results. Thus, we confirmed the structural integrity of the RCCV at design pressure.

Leak tests after S1 and S2 excitation tests confirmed that the RCCV's leak tightness was maintained under S1 and S2 motions, which verified the leak tightness function of the liner system.

Simulation analysis results of a 3-dimensional nonlinear FEM model for the pressure test, S1(H) and S2(H) verification tests gave good agreement with measured structural behaviors.

Failure test was conducted to obtain the seismic safety margin of the RCCV. The maximum input acceleration was about 9 times that of S2 and functional soundness of the specimen was maintained up to the failure. So, seismic margin of the specimen is supposed more than 5 times the S2.

ACKNOWLEDGMENT

Since 1980, NUPEC has been conducting a series of seismic proving tests of nuclear power plant facilities under the sponsorship of the Ministry of International Trade and Industry(MITI) of Japan. The seismic proving test of the RCCV (and PCCV) has been carried out as one of the seismic proving tests. The authors would like to acknowledge the advice of the Seismic Proving Test Executive Committee and CCV's Subcommittee of NUPEC.

REFERENCES

Nakamura,S.,Sasaki,Y.,Horibe,K.,Nakajima,M.,Hasegawa,T.,Kei,T.,Mieda,T.,Takiguchi, K. and Akiyama,H. 1997 "Plan of the Seismic Proving Test for Reinforced Concrete Containmnent Vessel, Part-1: Test Plan and Test Model Design" SMiRT 14, Vol.K. Lyon,France,August 1997.

Hattori,K.,Tsurumaki,S.,Sasaki,Y.,Akiyama,H.,Takiguchi,K.,Satoh,K.,Eto,H.,Goto,M. and Kobayashi,T. 1998 "Seismic Proving Test of Concrete Containment Vessels" 7th International Symposium on Current Issues Related to Nuclear Power Plant Structures, Equipment and Piping, SessionIV. Raleigh, North Carolina, December 1998.

Hattori,K.,Tateno,S.,Katayama,H.,Naito,Y. and Akiyama,H. 1999 "Seismic Proving Test of a Reinforced Concrete Containment Vessel (RCCV), Part 1: Outline of the Test, Result from Small Acceleration Excitation and Analysis Method for Observed Data" ASME/PVP Session 1.1A. Boston, August 1999.

Sasaki,Y., Tsurumaki,S., Hirama,T., Yamanaka,H. and Akiyama,H. 1999 "Seismic Proving Test of a Reinforced Concrete Containment Vessel (RCCV), Part 2: Test Results and Simulation Analyses" ASME/PVP Session 1.1A. Boston, August 1999.

Tateno,S.,Hattori,K.,Shiba, K.,Kei,T. and Akiyama,H. 1999 "Seismic Proving Test of a Reinforced Concrete Containment Vessel (RCCV), Part 2: Results of Shaking Table Tests" SMiRT 15, Vol.K. Seoul, Korea, August 1999.

Tsurumaki,S., Tateno,S., Sasaki,Y., Kobayashi,T. and Nakajima,M. 1999 "Seismic Proving Test of a Reinforced Concrete Containment Vessel (RCCV), Part 3: Simulation Analysis of Shaking Table Tests" SMiRT 15, Vol.K. Seoul, Korea, August 1999.

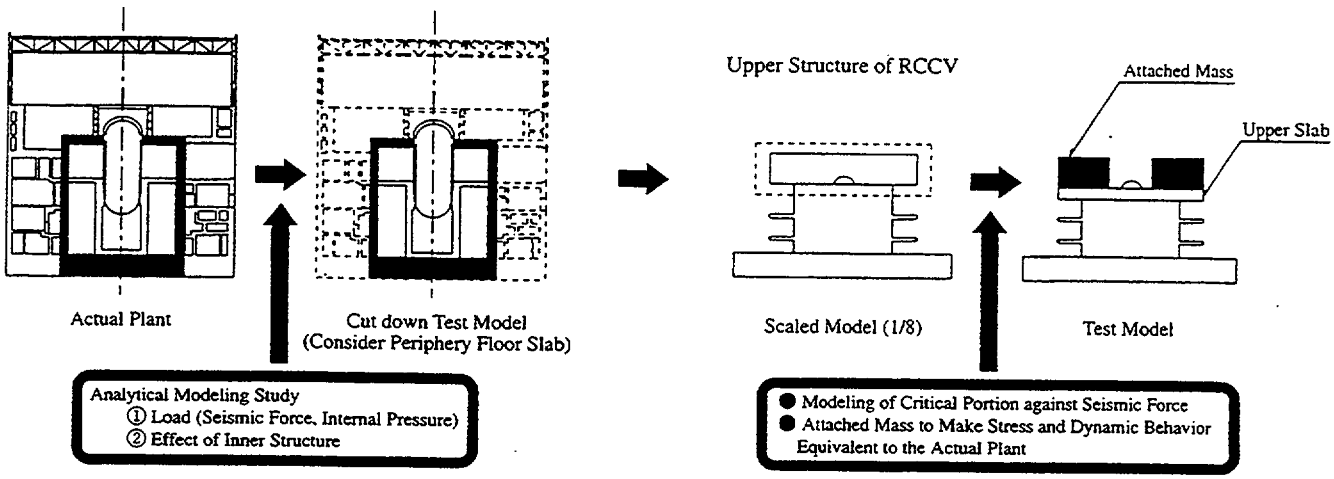


Fig.1 Modeling Procedure

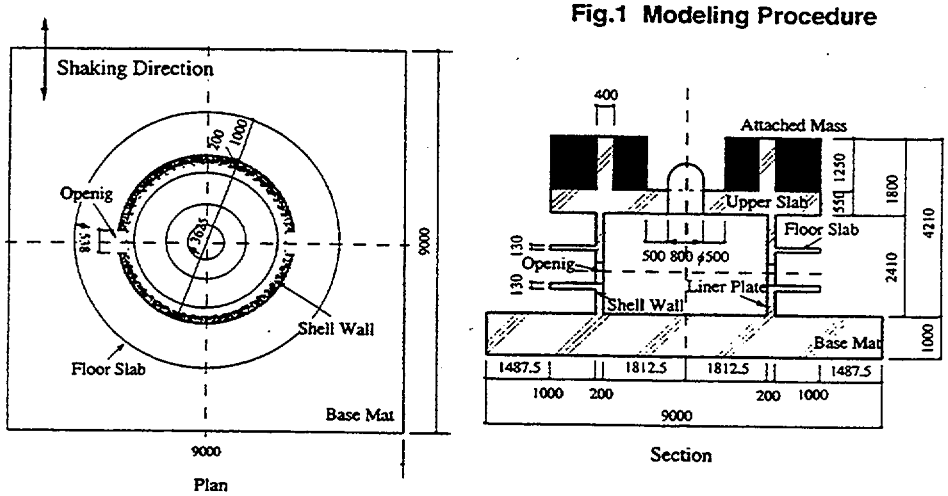


Fig. 2 Test Model Configuration

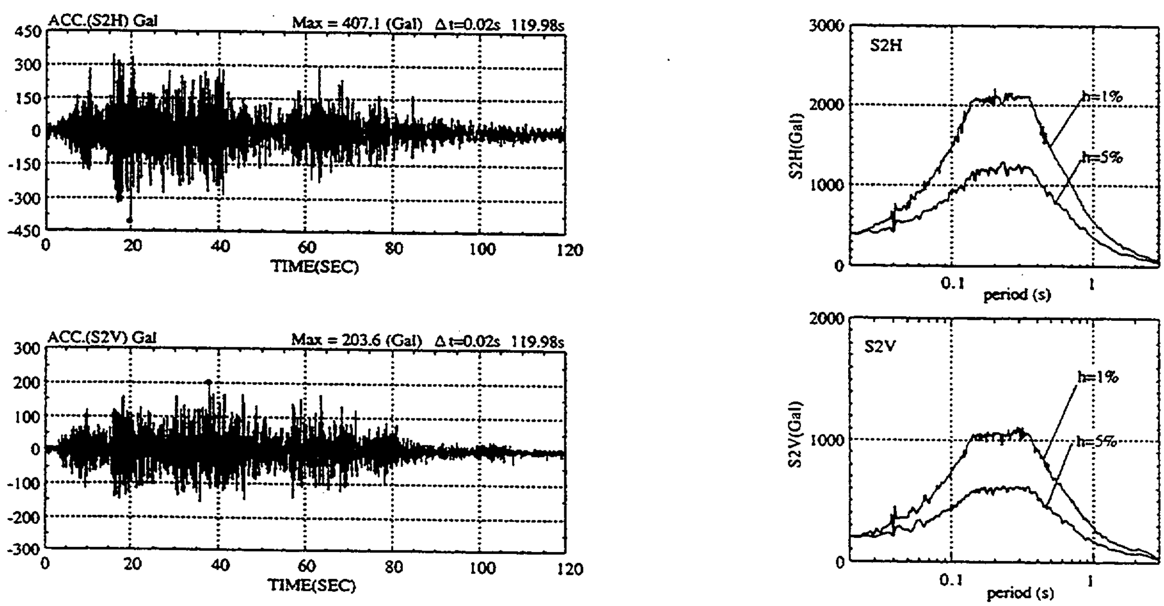
Table 1 Outline of the Model

(a) Test Model Scale

Whole Test Model Scale	1/8 (ID. 3625mm)
Concrete Wall Thickness	1/10 (200mm)
Liner Plate Thickness	1/4 (1.6mm)

(b) Test Model Weight (ton)

Base Mat	213
ShellWall and Slabs	76
Attached Mass	276
Measuring Frame and Protection Equipment	30
Total	595



(a) wave form

(b) response spectrum

Fig.3 Input Earthquake

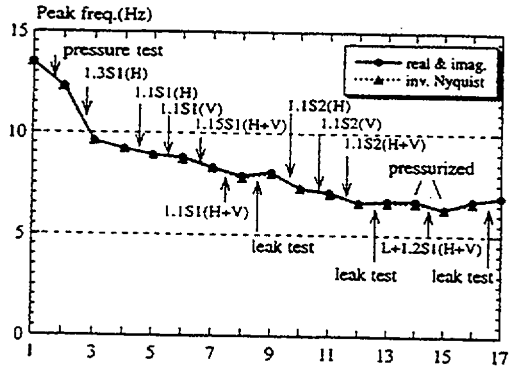


Fig.4 Transition of the 1st Peak Frequency

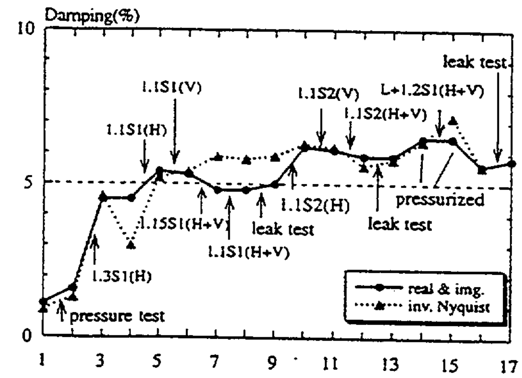


Fig.5 Transition of the 1st Peak Damping Ratio

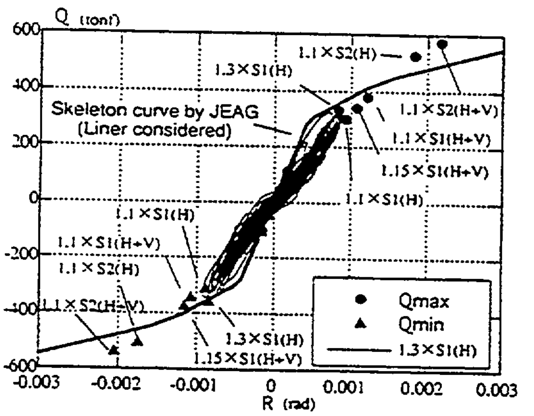


Fig.6 Maximum Value of Base Shear and Deformation Angle

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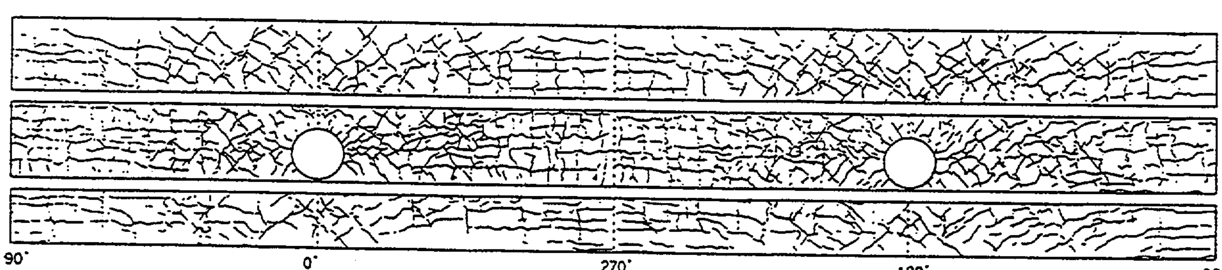


Fig.7 Cracks Observed in Leak Test (after 1.1S2(H+V) Excitation)

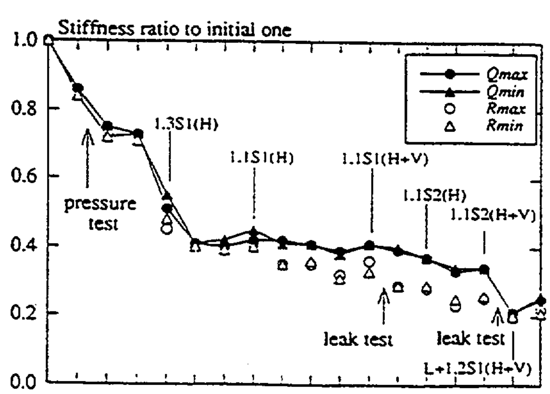


Fig.8 Transition in Stiffness Deterioration Rate

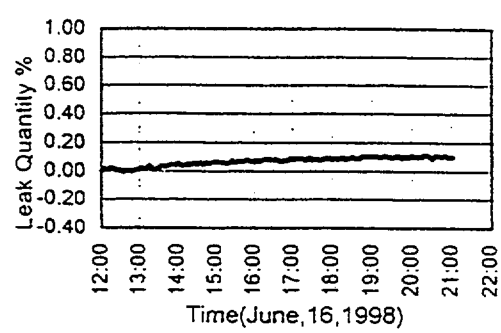


Fig.9 Change of Leak Quantity

Table 2 Material Properties used in Simulation Analysis

Concrete (Fc330)			Rebar (SD390)		Liner (SM490A)	
Ec kgf/cm^2	fc kgf/cm^2	ft kgf/cm^2	Es kgf/cm^2	σ y kgf/cm^2	Es kgf/cm^2	σ y kgf/cm^2
2.1×10^5	328	17	2.1×10^6	4700	2.1×10^6	3430

Ec, Es : Young's Modulus
fc : Compressive Strength of Concrete
ft : Tensile Strength
σ y : Yielding Strength

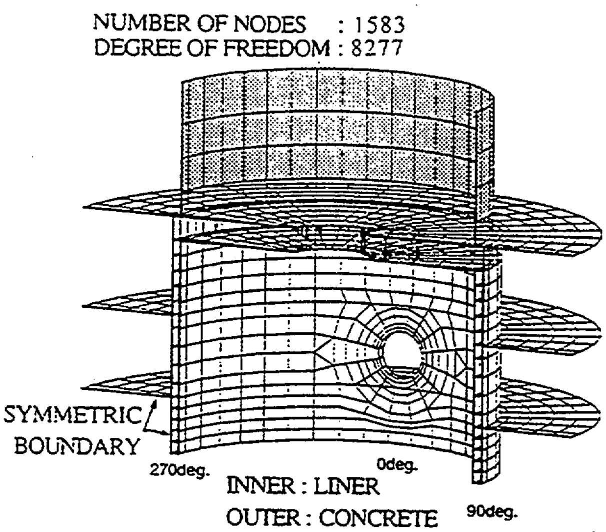
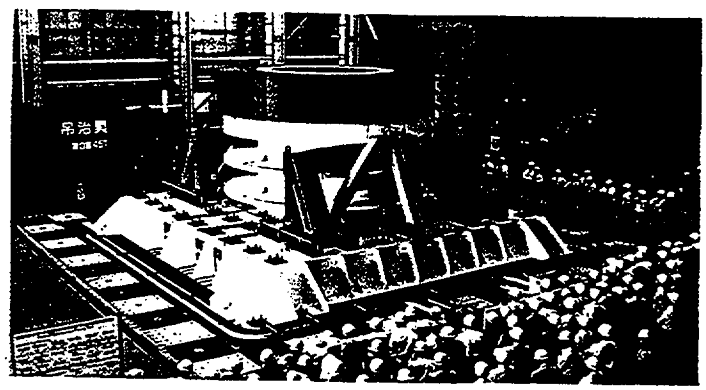


Fig. 10 3-Dimensional Finite Element Model



Demonstration Test for Public

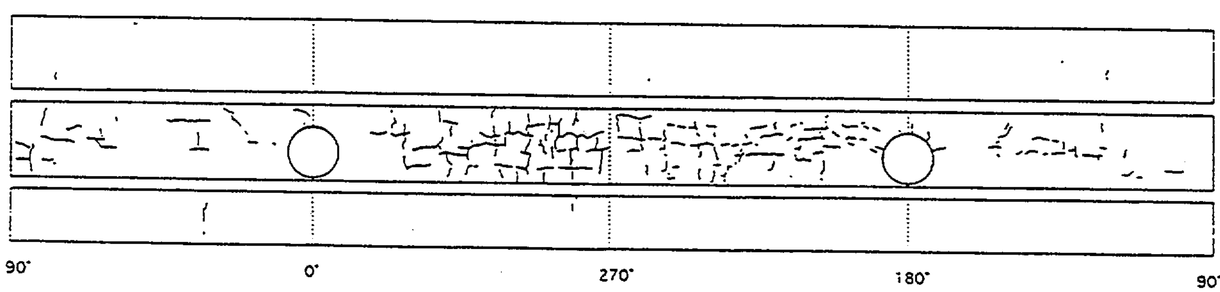
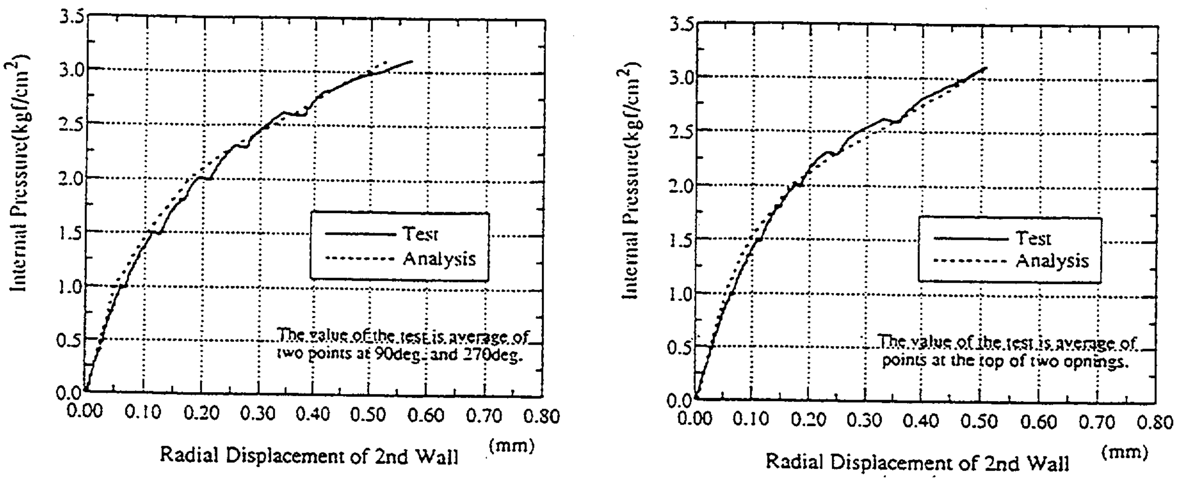


Fig.11 Cracks Observed in Pressure Test



(1) At 90deg. and 270deg.

(2) At the Top of Opening

Fig.12 Radial Displacement of the 2nd Story

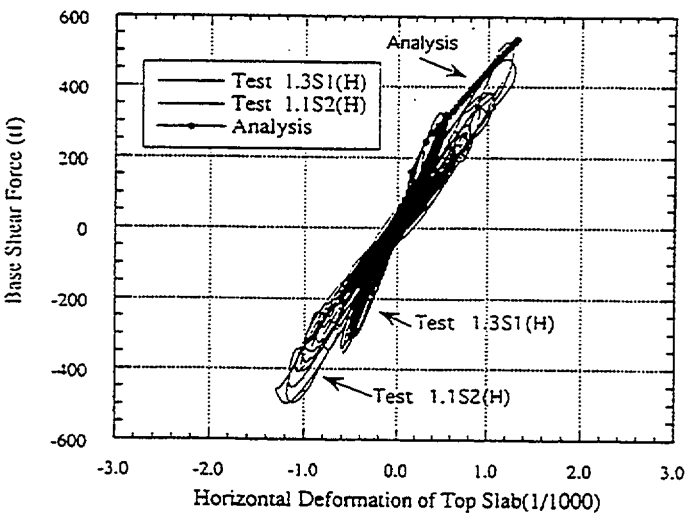


Fig.13 Base Shear Force - Horizontal Deformation Angle at the Top Slab

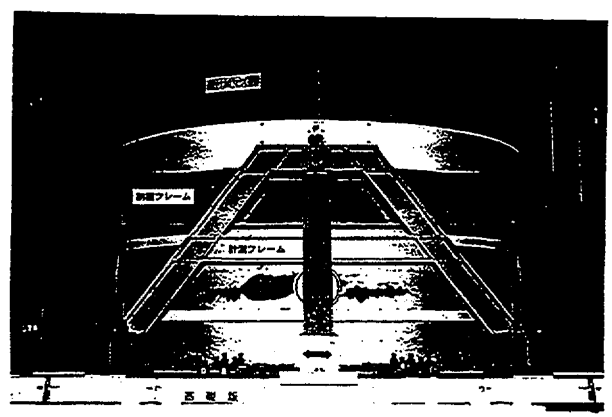


Photo1 Failure state

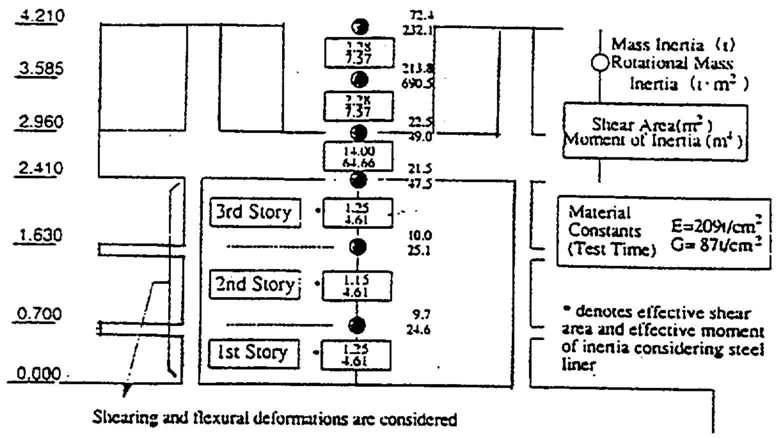


Fig.14 Analysis Model for Horizontal

Table 3 Analysis Cases(Simulation Model 1)

Number	Simulated Test	Analysis Method	Input Motion	Skeleton Curve	Hysteresis Loop
1	1.3×S1(H)	Time History Nonlinear	Horizontal + Pitching	JEAG + Liner Effect	INADA Model
2	1.1×S2(H)	Time History Nonlinear	Horizontal + Pitching	JEAG + Liner Effect	INADA Model
3	1.1× S2(H+V)	Time History Nonlinear	Horizontal + Pitching Equivalent Horizontal	JEAG + Liner Effect	INADA Model

Horizontal : $\ddot{u}_0$, Pitching : $\ddot{\theta}_0$, Equivalent Horizontal : $\ddot{u}_0 + heq\ddot{\theta}_0$

Table 4 Transfer Function

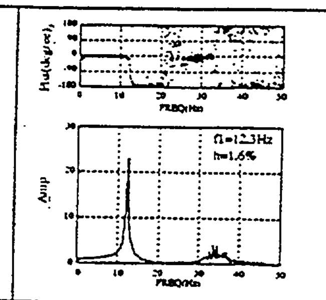
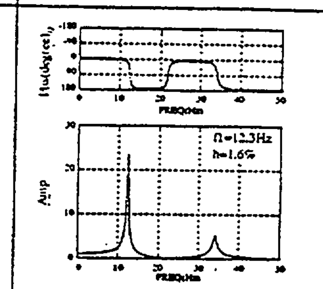
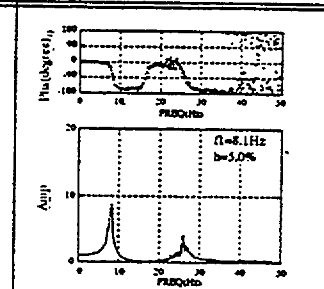
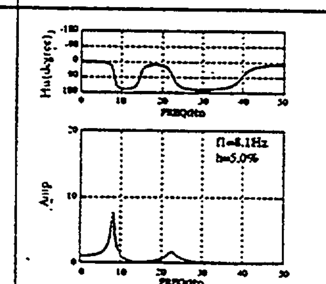
Test Specimen Characteristic Test Before 1.3 x S1(H) Excitation	
Analysis Before 1.3 x S1(H) Excitation	
Test Specimen Characteristic Test Before 1.1 x S2(H) Excitation	
Analysis Before 1.1 x S2(H) Excitation	

Table 5 Comparison of Time Histories of Horizontal Acceleration at Top Slab

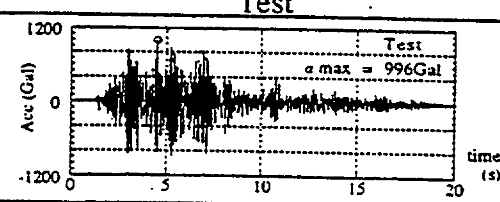
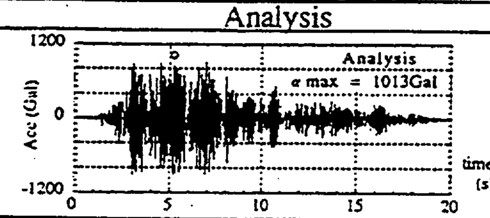
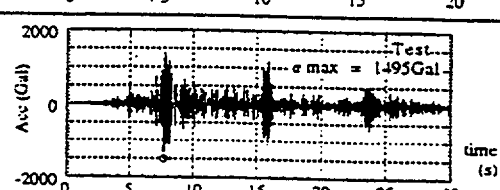
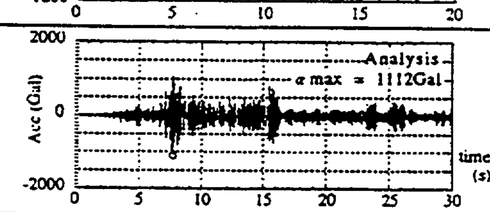
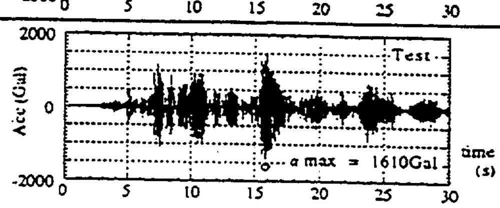
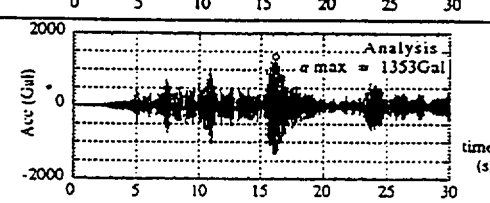
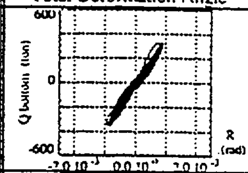
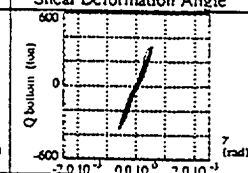
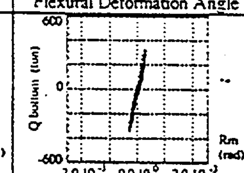
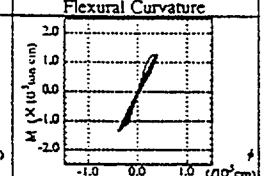
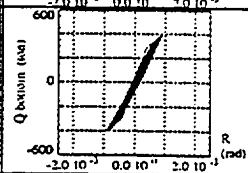
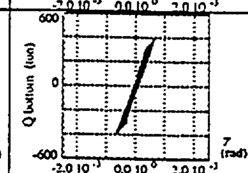
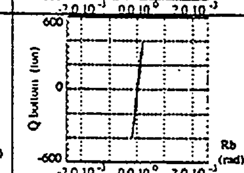
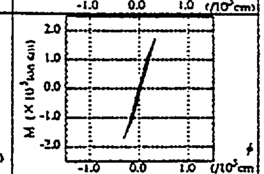
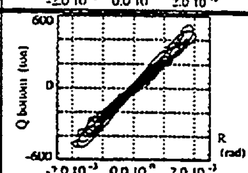
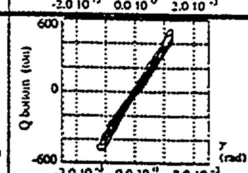
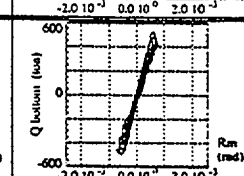
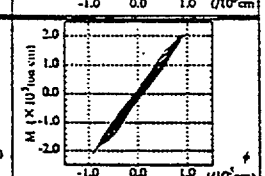
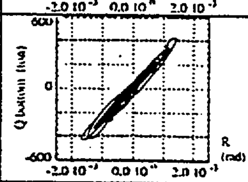
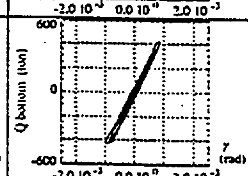
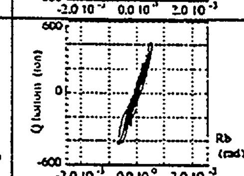
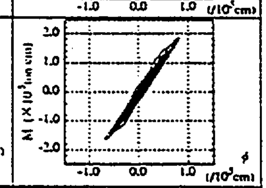
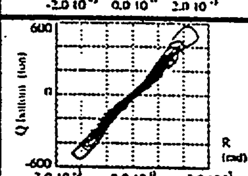
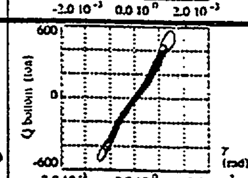
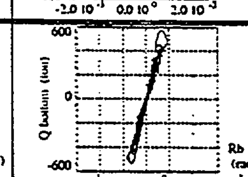
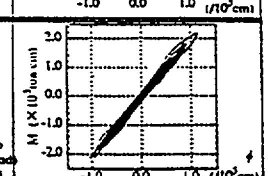
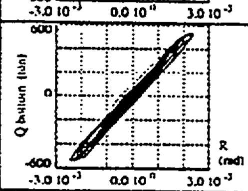
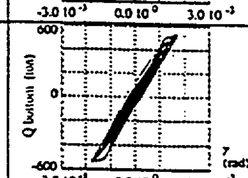
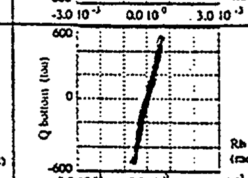
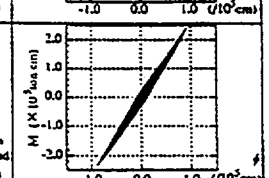
	Test	Analysis
1.3× S1(H)		
1.1× S2(H)		
1.1× S2(H+V)		

Table 6 Comparison of Hysteresis Loops between Forces and Deformation

	Q−R Shear Force - Total Deformation Angle	Q−γ Shear Force - Shear Deformation Angle	Q−Rb Shear Force - Flexural Deformation Angle	M−ϕ Bending Moment - Flexural Curvature
1.3× S1(H) Test Result				
1.3× S1(H) Analysis Result				
1.1× S2(H) Test Result				
1.1× S2(H) Analysis Result				
1.1× S2(H+V) Test Result				
1.1× S2(H+V) Analysis Result				

JAERI RESEARCH ON FUEL ROD BEHAVIOR DURING ACCIDENT CONDITIONS

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This paper describes an outline of LOCA test program, and presents recent results from RIA experiments in the NSRR and related mechanical testing, i.e. tube burst test for hydrided cladding. The NSRR experiments with high burnup BWR fuels and ATR/MOX fuels provide information regarding rod deformation and fission gas release during the RIA transient. The tube burst test shows the effect of hydride on cladding failure at room temperature at an elevated temperature.

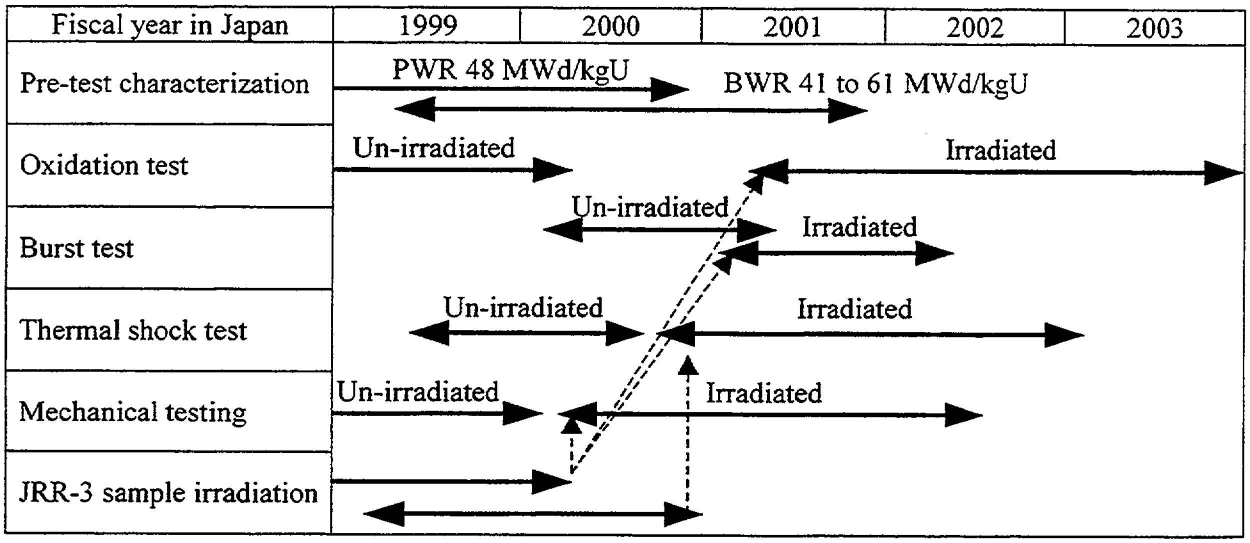
I. INTRODUCTION

To provide a database for the regulatory guide of light water reactors, behavior of reactor fuels during accident conditions is being extensively studied in the Japan Atomic Energy Research Institute (JAERI). The research activities cover fuel behavior during reactivity-initiated accident (RIA), loss-of-coolant accident (LOCA), thermal-hydraulic power oscillation and severe accident. This paper gives an outline of LOCA test program, and describes most recent results from RIA experiments and related mechanical testing for high burnup cladding.

II. LOCA TEST PROGRAM

The objectives of the LOCA test program are to evaluate the influence of high burnup effects on fuel behavior under LOCA conditions, and to provide basic data to assess applicability of the current criteria, 1200 deg C or 2200 F, 15%ECR (Equivalent cladding reacted, proportion of oxide layer thickness assuming that all of absorbed oxygen forms stoichiometric ZrO_2). The program comprises; (1) Pre-test characterization of cladding through extensive PIEs (post-irradiation examinations) to clarify high burnup effects which are likely to influence fuel behavior under LOCA conditions; (2) LOCA tests including; (2a) Oxidation test to quantify effects of pre-oxidation and pre-hydriding separately; (2b) Tube burst test; (2c) Integral thermal shock test to determine failure-bearing capability; (2d) mechanical testing to quantify mechanical properties of cladding which experienced temperature transient; and (3) Computer code simulation to predict behavior of fuel bundle under LOCA conditions. In the experiments, simulated high burnup fuel claddings are used as well as cladding sampled from fuel rod irradiated in commercial power-producing reactors. The simulated high burnup fuel claddings are pre-oxidized and/or pre-hydrided un-irradiated claddings and pre-hydrided samples irradiated in JAERI's research reactor JRR-3. These samples realize study to quantify separate effects of pre-oxidation and pre-hydriding, and a combination of these effects. The schedule of the LOCA test program is shown in Table 1.

Table 1 Schedule of LOCA test program at JAERI



III. RIA TEST PROGRAM

The objectives of the RIA test program are; to investigate fuel behavior of high burnup fuel rod under the simulated RIA conditions; to determine the fuel rod failure threshold of irradiated fuel rod and to evaluate influences of the fuel burnup; and to clarify the modes, mechanisms and consequences of the fuel failure. The program is composed of pulse-irradiation experiments in the NSRR (Nuclear Safety Research Reactor), related separate-effect tests and code simulations. As integral tests in the NSRR, number of irradiated PWR, BWR and ATR/MOX fuels are subjected to the pulse-irradiations. In addition to the irradiated fuel tests, non-constraint irradiated pellet slice, fine particle fuel and un-irradiated fuels with pre-hydrided and/or pre-oxidized cladding are also used in analytical experiments to investigate potential of fuel pellet fission-gas-induced expansion, thermal interaction of solid fuel particles with coolant water, and effects of cladding pre-oxidation and pre-hydriding. As a part of the NSRR program, significant activities are continued to develop high-temperature and high-pressure test capsule, transient swelling sensor with eddy current detection, and video system for in si-tu observation (already existing for un-irradiated fuels, but to be developed for irradiated fuels). Currently particular emphasis is in the calibration and noise reduction of the transient swelling sensor, since the sensor can provide data of cladding expansion in early phase, PCMI loading process, and key information regarding a possible contribution of fission-gas-induced pellet expansion to the PCMI loading.

In addition to the NSRR experiments, cladding mechanical properties are investigated through tube burst test and modified tensile test with machined ring specimen. As for fission gas release from high burnup fuels, fuel heating experiments with newly installed VEGA and OGA facilities are continued. Code assessment and model development are performed with FRAP-T6 and FRAPTRAN codes from USNRC and SCANAIR code from French IPSN.

III.1. BWR Fuel Test

In addition to the extensive efforts on high burnup PWR fuel experiments[1-4], high burnup BWR fuels (41 to 60 MWd/kgU) are being tested in the FK test series[5] in the NSRR. We have performed two experiments, FK-4 and FK-5, with 56 MWd/kgU fuels in early 1999. In terms of relative importance of RIA fuel behavior study, a priority of PWR fuel experiments to BWR fuel tests is recognized due to the lower probability of rod drop accident in BWR. However, the high burnup BWR tests give important information to promote a better understanding of ruling phenomena in RIA fuel behavior, and provide useful data on possible fuel behavior during anticipated transient without scram (ATWS) in BWR. Test conditions of the FK test series are listed in Table 2. 8x8BJ Step I fuels irradiated for five cycles in Fukushima I-3 reactor are subjected to the Tests FK-1 through -3, and 8x8 Step II fuels irradiated for four cycles in Fukushima II-2 reactor are tested in the Tests FK-4 and -5. Since peak linear heat rate during base-irradiations in the BWRs varies from 19.3 kW/m to 35.2 kW/m, fission gas release during the base-irradiations are significantly different among the fuels. Fuel burnup of the test rods are from 41 MWd/kgU to 56 MWd/kgU, and fuel enthalpy exceeds 140 cal/g during pulse-irradiations of the Tests FK-3 and -4. Fuel failure did not occur in these FK tests. Tests FK-6 and -7 are to be performed with 8x8 Step II fuels irradiated for five cycles in Fukushima II-2 reactor in February and March, 2000.

Table 2 Test conditions of BWR/FK experiments

Test ID	Peak linear heat rate (kW/m)	Fuel burnup (MWd/kgU)	FGR during base-irradiation (%)	Fill gas pressure of test rod (MPa)	Peak fuel enthalpy during pulse (cal/g)
FK-1	22.8	45	1.5	0.3	130
FK-2	22.8	45	1.5	0.3	70
FK-3	19.3	41	0.35	0.3	145
FK-4	35.2	56	12.5	0.5	~140
FK-5	35.2	56	12.5	0.5	~70

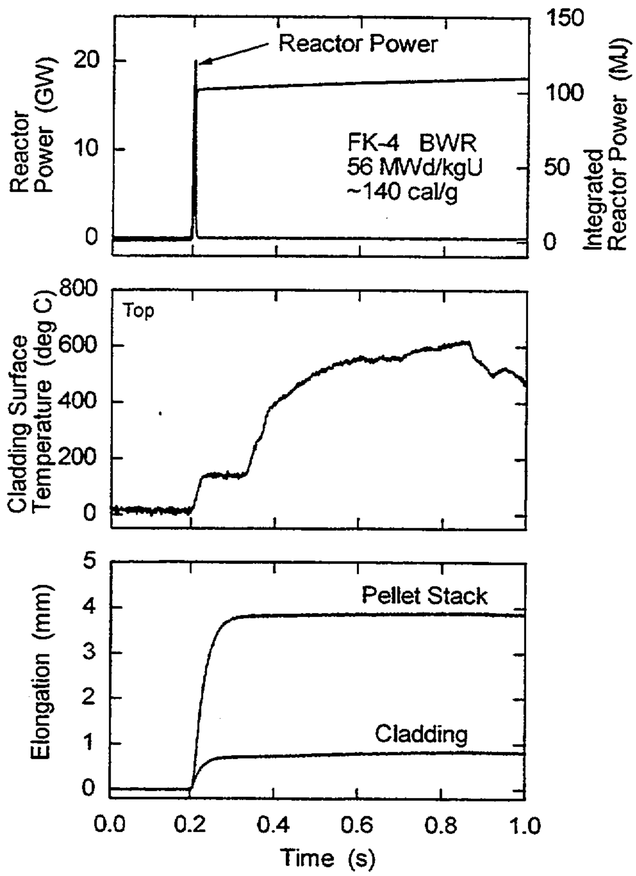


Fig. 1 Transient records in the Test FK-4

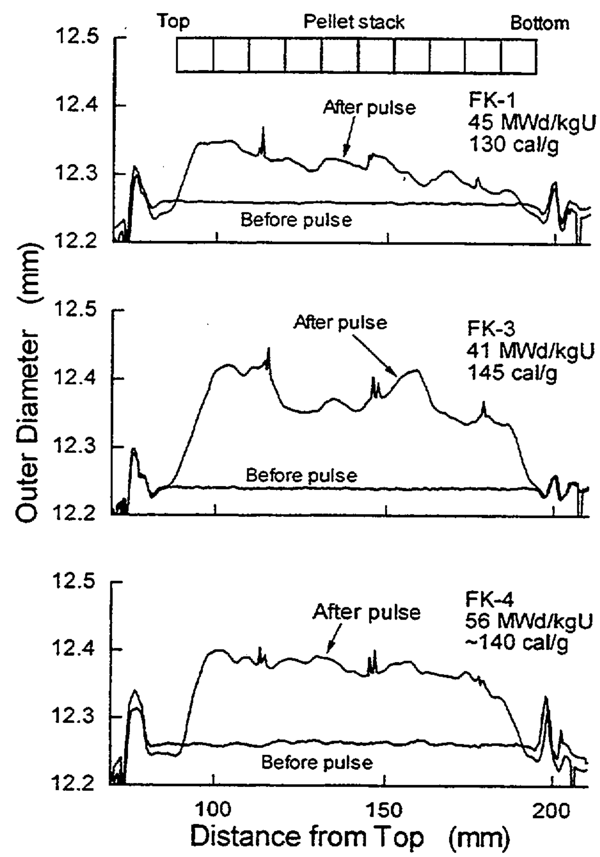


Fig. 2 Axial profile of cladding outer diameter

Figure 1 shows transient histories of reactor power, cladding surface temperature, and axial elongation of cladding and pellet stack during the Test FK-4. Cladding surface temperature increases rapidly at pulse, and reached 600 deg C at maximum. Pellet stack and cladding elongations are about 4 mm and 1 mm, respectively. Fuel deformation was not significant in the Test FK-2 and -5 due to the low enthalpy level, i.e. ~70 cal/g, but permanent deformation occurred in the Tests FK-1, -3 and -5 with the higher enthalpy during transients. Axial profile of cladding outer diameter before and after pulse-irradiations of the Tests FK-1, -3 and -4 are shown in Fig. 2. Cladding outer diameter increased in these tests, and the largest increase was in the FK-3 with the highest fuel enthalpy among the three tests. Although fuel burnup of the FK-4 (56 MWd/kgU) is much higher than that of the FK-3 (41 MWd/kgU) and peak fuel enthalpy is in the same level (~140 cal/g for the FK-4 and 145 cal/g for the FK-3), the deformation is smaller in the FK-4 than in the FK-3. As seen in Figure 3, residual hoop strain of the post-test cladding is most dependent on peak fuel enthalpy in the BWR fuel tests. Data from TS test series[6] with 26 MWd/kgU BWR fuels and from PWR fuels[3] are also plotted in the figure. Fuel swelling of BWR fuels is less significant than that of PWR fuels and residual strain appears only in the higher enthalpy level. Since creep down of cladding is less significant in BWRs comparing with that in PWRs, wider gap between fuel pellet and cladding inner surface (P/C gap) exists in BWR fuels before pulse-irradiation in the NSRR. The wider pre-test gap provides BWR fuels with weaker PCMI loading during the transients, and reflects smaller fuel rod deformation during the transient.

The BWR fuel experiments give characteristic nature in fission gas release during transients. Figure 4 shows fission gas release as a function of peak fuel enthalpy. The results indicate that fission gas release in the BWR fuels during pulse irradiation is strongly influenced by base-irradiation conditions, e.g. linear heat rate. Fuel with the larger fission gas release during base-irradiation gives the larger fission gas release during transient. This indicates that release paths generated during base-irradiation play a role also during transient.

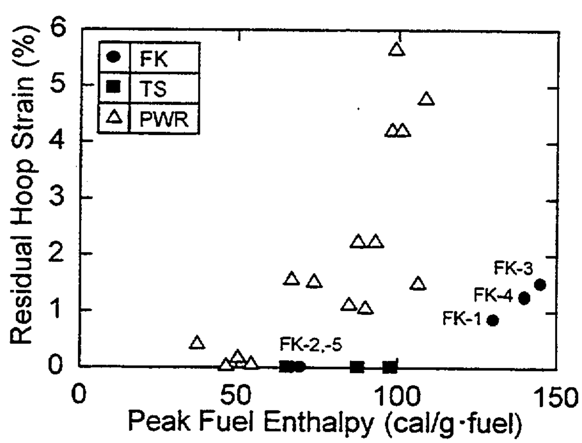


Fig. 3 Residual hoop strain (BWR/FK)

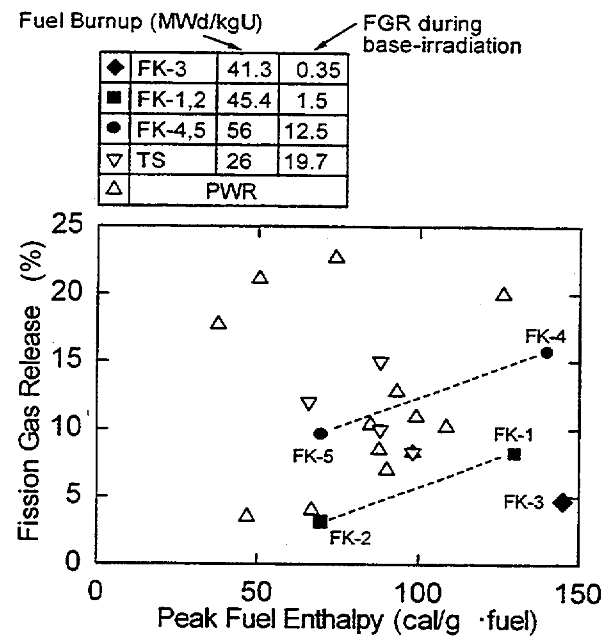


Fig. 4 Fission gas release (BWR/FK)

III.2. ATR/MOX Fuel Test

The test fuel rod used in the ATR/MOX series[7] is sampled from the ATR 36-type MOX fuel rod irradiated in the prototype advanced thermal reactor (ATR) 'Fugen' of Japan Nuclear Cycle Development Institute (JNC). The fuel has initial plutonium enrichment of 2.98% in Pu-239 and Pu-241, and has been base-irradiated with peak linear heat rate of 23.5 kW/m. Burnup reached approximately 20 MWd/kgHM. During the base-irradiation, oxide thickness of 7 to 15 μm forms on cladding outer surface. Released fission gas at the end of the base-irradiation is 0.2% or lower of total inventory. The four test rods in the Tests ATR-1 through -4 were cut from the eighth, seventh, fifth and ninth spans of the mother rod, respectively. As-fabricated fuel pellet is 12.4 mm in diameter and 13 mm in height, and the pellets are sheathed with Zircaloy-2 cladding of 14.5 mm in outer diameter and 0.9 mm in thickness. Accordingly, initial P/C gap is 150 μm. The P/C gap reduced to 75 to 80 μm due to creep down and/or pellet swelling during the base-irradiation. Each short-sized test rod contains nine fuel pellets, and has total length of about 300 mm including fuel stack region of about 120 mm. The test rod is filled with pure helium of about 0.3 MPa simulating gap gas condition at the end of the base-irradiation. Peak fuel enthalpy during pulse, residual hoop strain of post-test cladding and fission gas release during the transient of the four experiments are listed in Table 3.

Table 3 Matrix of ATR/MOX tests

Test ID	Peak fuel enthalpy (cal/g)	Residual hoop strain (%)	Fission gas release during transient (%)
ATR-1	80	0	1.0
ATR-2	110	0	8.8
ATR-3	120	1.9	17.7
ATR-4	140	3.2	19.9

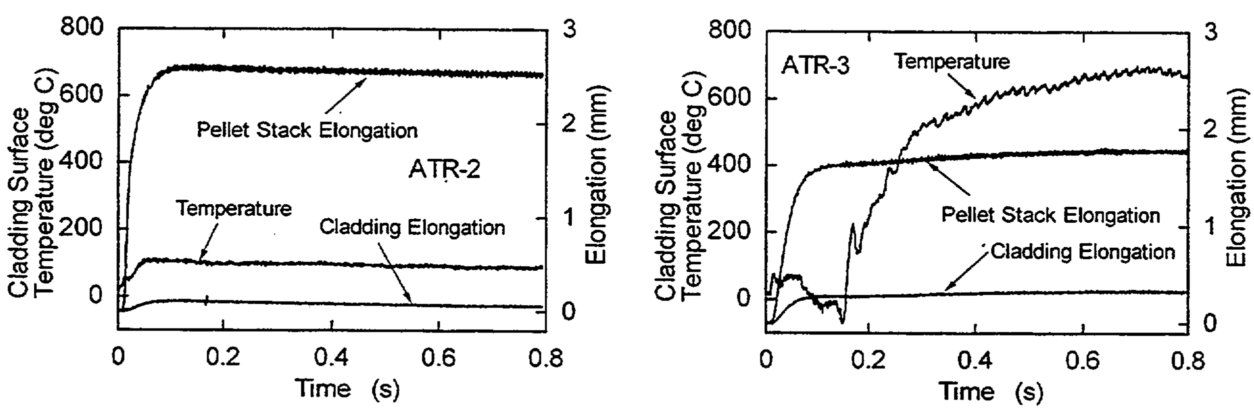


Fig. 5 Transient records in the Tests ATR-2 and ATR-3

Figures 5 compares transient records of cladding surface temperature, pellet stack and cladding elongation between the Tests ATR-2 and -3. Peak fuel enthalpy is 110 cal/g in the ATR-2, and is 120 cal/g in the ATR-3. The lower cladding temperature, larger stack elongation and smaller cladding elongation in the Test ATR-2 suggest that P/C gap remains open during the transient. On the other hand, data of the Test ATR-3 show the higher cladding temperature, smaller stack elongation and larger cladding elongation, and indicate occurrence of gap closure and resulting PCMI loading. Hence, threshold of PCMI in terms of peak fuel enthalpy exists between 110 to 120 cal/g. Radial cross-sections of post-test fuels are shown in Fig. 6. PCMI loading and compressive stress in fuel pellet cause fine radial cracks in fuel pellet periphery and, in particular, circumferential cracks in the Tests ATR-4 with the higher peak fuel enthalpy of 140 cal/g. Figure 7 shows residual hoop strain of post-test ATR cladding as a function of peak fuel enthalpy. The ATR/MOX fuels have a pre-test state similar to BWR/FK fuels in terms of cladding material and P/C gap, and fuel burnup of the ATR/MOX fuels are much lower than that of the BWR/FK fuels. Residual strain in the ATR/MOX fuels is, however, significantly larger than that in the FK test series.

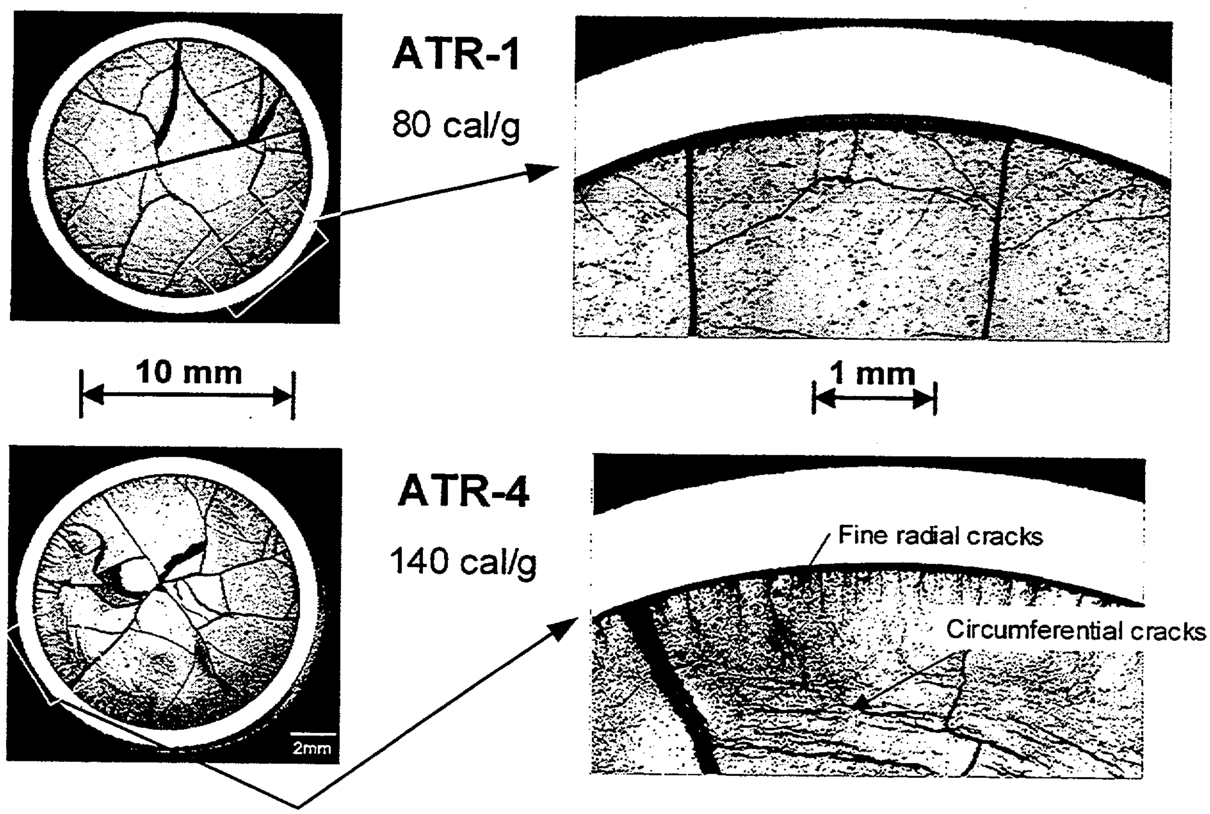


Fig. 6 Radial cross-sections of post-test fuels.

The ATR/MOX fuels give high values also for fission gas release during transient. Fission gas release in the ATR/MOX tests is plotted in Fig. 8 as function of enthalpy increase (equivalent to peak fuel enthalpy in the NSRR experiment). Data from the BWR/FK and from MOX fuel experiments in Cabri REP-Na program[8] are also shown in the figure. Fission gas release in the MOX fuel experiments including two Cabri tests Na-6 and Na-9[9] well correlates with enthalpy increase, and larger enthalpy increase results in larger fission gas release. Although fuel burnup of the tested ATR fuels are limited, fission gas release in the Tests ATR-4 reaches 19.9%. Figure 9 shows microstructures observed in radial cross-section of fuel pellet sibling sample, which is not experienced the transient. Plutonium agglomerates (Pu spot) in diameter of 10 to 40 μm can be seen in extensive area of the pellet. Finer grain structure in the vicinity of the Pu spot is observed in peripheral region, and is similar to rim structure observed in high burnup UO_2 fuel. This is due to extremely high local burnup in the Pu spot. Microstructures appeared in the post-test ATR-3 fuel are shown in Fig. 10. Large pores in Pu spot, micro-cracks initiated from the spot, grain boundary separation in the vicinity of the spot are observed in the post-test fuel. These appearances indicate a role of fission gas accumulated in the Pu spot on the crack initiation and grain boundary separation. These phenomena have a potential to cause the large fission gas release and the relatively large radial deformation of the ATR/MOX fuels during the transients.

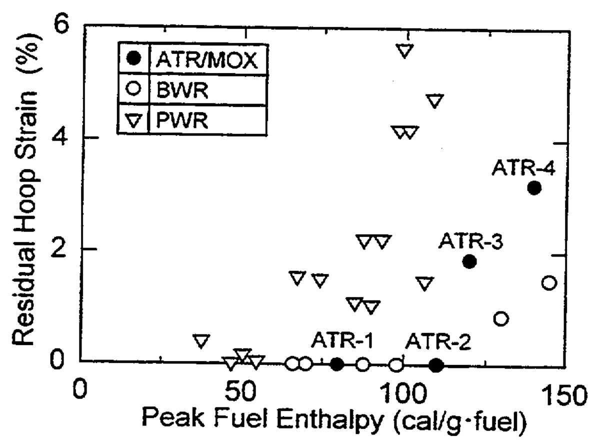


Fig. 7 Residual hoop strain (ATR/MOX)

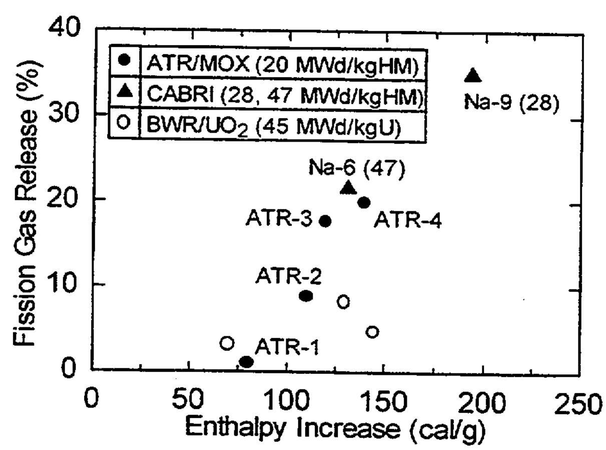


Fig. 8 Fission gas release (ATR/MOX)

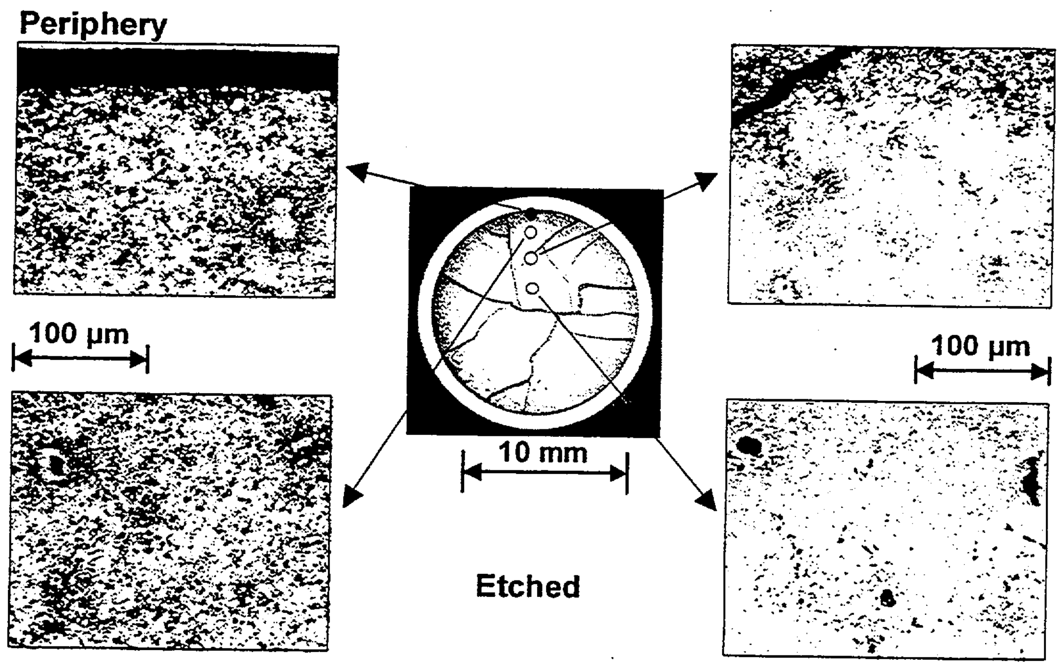


Fig. 9 Fuel micro-structure in sibling sample (without pulse)

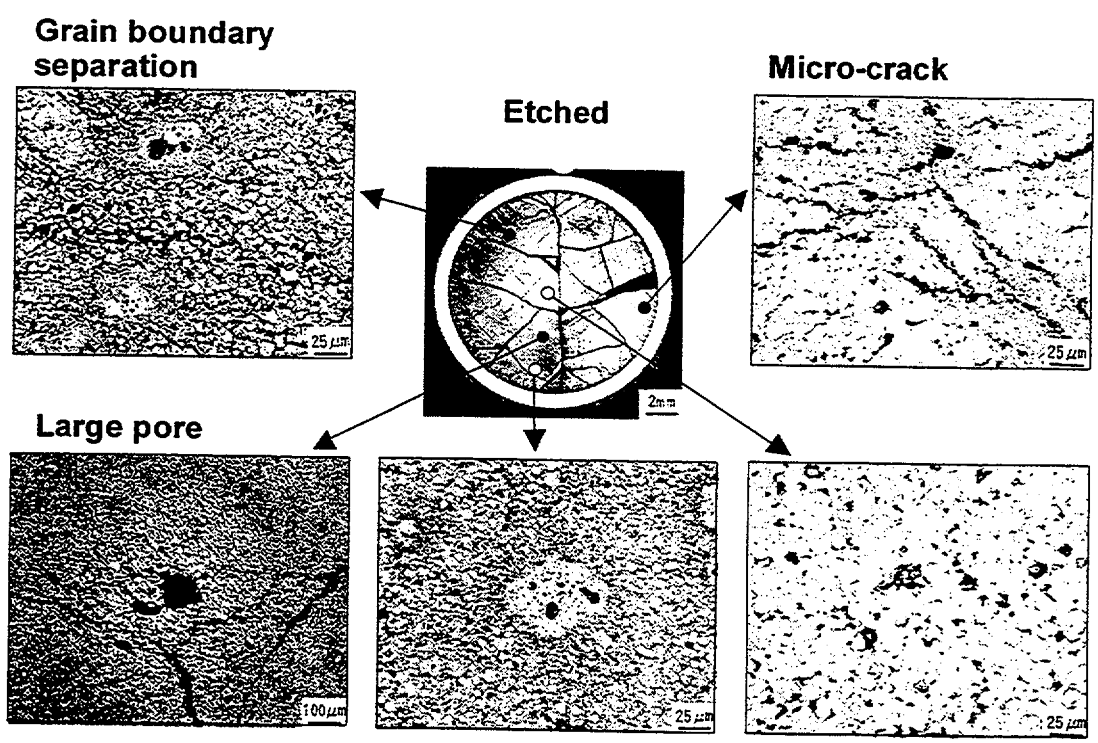


Fig. 10 Fuel micro-structure in post-test ATR-3 fuel

III.3. Tube Burst Test

Transient tube burst test has been performed simulating PCMI loading in the high burnup PWR fuel rod during an RIA. The objective of the test is to understand failure behavior of embrittled cladding, in particular, effect of hydride rim (radially-localized hydride layer) by using artificially hydrided cladding sample. The test set-up and results from tests at room temperature are described in previous documents.[10,11] Samples with hydride rim were tested at an elevated temperature (620 K) subsequently to the tests at room temperature. The cladding sample was heated to 620 K at a rate of 1 K/s with an infrared furnace and pressurization was initiated after 900 s duration. Sample temperature is measured with K type thermocouple attached to the surface. Radial temperature in the sample is nearly uniform. On the other hand, the cladding temperature has an axial profile, and the profile has a peak at 60 mm below the top end. The temperature difference between the peak position and the top end is within 10 K, while the temperature decreases continuously to the bottom end and the temperature difference between the peak position and the bottom end is about 50 K. The temperature at the peak position is maintained at 620 K during the experiment.

Test specimens are low-tin (1.3wt%Sn) Zircaloy-4 cladding tubes. The inner and outer diameters of the sample are 9.50 mm and 8.36 mm, respectively, and the length is 160 mm. The tests were conducted on three types of specimens consisting as-received sample (non-hydrided); uniformly hydrided sample; and hydrided sample with hydride rim. Hydrogenation was performed in mixture gas of hydrogen and argon at about 600 K, and samples with hydrogen concentrations of 100 to 1100 wtppm were produced. Variation in hydrogen concentration throughout a sample was estimated to be within ±30% from the analysis of reference sample. In the sample with hydride rim, hydrides are localized in 50 to 150 μm of sample periphery. To evaluate local hydrogen concentrations inside and outside the hydride rim, hydrogen content was measured in sample with mechanical removal of the rim. Hydrogen concentration outside the rim is 100 to 200 wtppm. Accordingly, hydrogen concentration inside the rim can be estimated from the concentration outside the rim, rim thickness, and the average of the sample. Local hydrogen concentration inside the hydride rim increases with increase of average concentration of sample, and varies from 2000 to 3000 wtppm. The hydrogen concentration inside the rim is about three times higher than the average of the sample.

Figure 11 shows post-test appearances of three different samples tested at room temperature and the sample with hydride rim tested at 620 K. At room temperature, non-hydrided sample exhibits a typical burst opening of 30 mm long, while axially extended openings are formed in the hydrided samples. In the sample with hydride rim tested at 620 K, a long axial crack as seen at room temperature is not formed. Figure 12 shows transient histories of sample internal pressure measured in the test at 620 K. Average hydrogen concentrations of the samples with hydride rim are 455, 602 and 1033 wtppm. They ruptured 150 to 280 ms after the onset of pressurization, obviously at earlier than the non-hydrided samples. Burst pressure of the samples with hydride rim is slightly lower than the non-hydrided samples.

Through photo-image analysis on radial cross-section at the axial position where the maximum deformation occurred in post-test sample, residual hoop strain was evaluated at both the inner and the outer surfaces. The residual hoop strain is shown in Fig. 13 as a function of hydrogen concentration.

Data obtained at room temperature are shown in Fig. 13a[10] and those at 620 K are in Fig. 13b. Larger hoop strain was always seen at the inner surface. The maximum and the minimum values of the error bar correspond to hoop strains at the inner and the outer surfaces, respectively. Hydrogen concentration was measured with hot extraction method for the sliced piece sampled from the vicinity of the position where the hoop strain was evaluated. In the tests at room temperature, as shown in Fig. 13a, the hoop strain decreases with the increase of hydrogen concentration, and the reduction becomes remarkable in hydrogen concentration of 300 wtppm or higher. The samples with hydride rim failed with significantly low hoop strain, less than 1% in average between the inner and outer surfaces. Figure 13b shows the residual hoop strain of the samples with hydride rim tested at 620 K. Thickness of hydride rim of each sample is indicated in the figure. The residual hoop strain is more than 10% in the cladding containing 250 wtppm of hydrogen (50 μm hydride rim), but is significantly reduced in samples with hydrogen concentration above 400 wtppm (with hydride rim thicker than about 100 μm). Even at 620 K, the residual hoop strain becomes less than 1% in samples with hydrogen concentration above 500 wtppm (with hydride rim thicker than 140 μm). Figure 14 shows radial cross-sections of failure cracks in two claddings with hydride rim. The sample with 50 μm hydride rim shows the hoop strain of 11.3%, but the sample with 140 μm hydride rim exhibits the strain of 0.8%. The sample with thinner hydride rim has wall thinning through the whole thickness which indicates enough ductility of the cladding and quite small influence of the hydride rim. On the other hand, the sample with thicker hydride rim shows a brittle fracture inside the rim, and the appearance of the crack is very similar to those observed in the Tests HBO-1 and TK-2 resulting in hydride-assisted PCMI failure of high burnup PWR fuels. The results from the tube burst test indicate an important role of the thicker hydride rim in the process of PCMI failure of the high burnup PWR fuels.

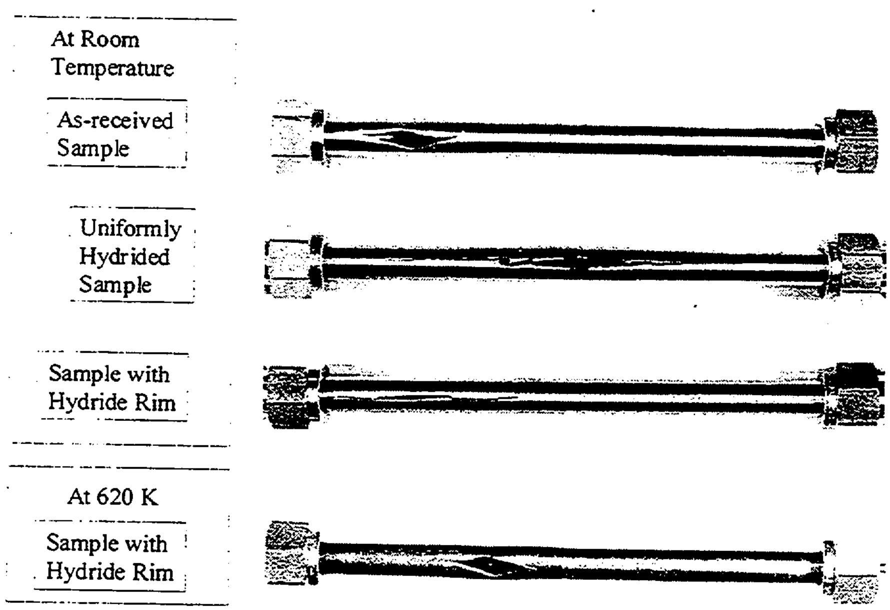


Fig. 11 Post-test appearances of samples in tube burst test

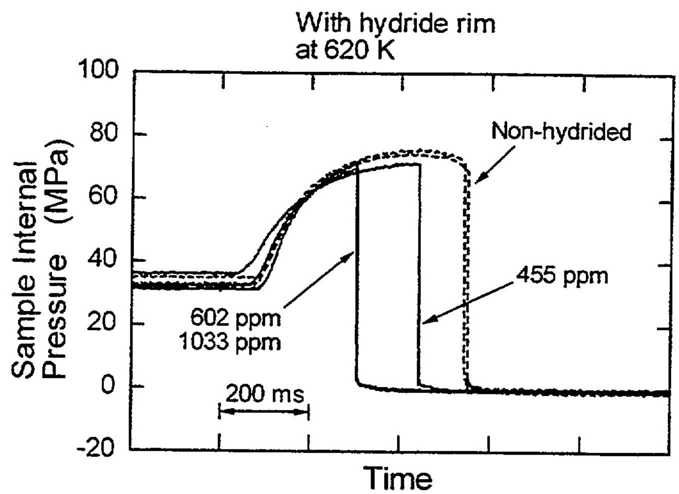
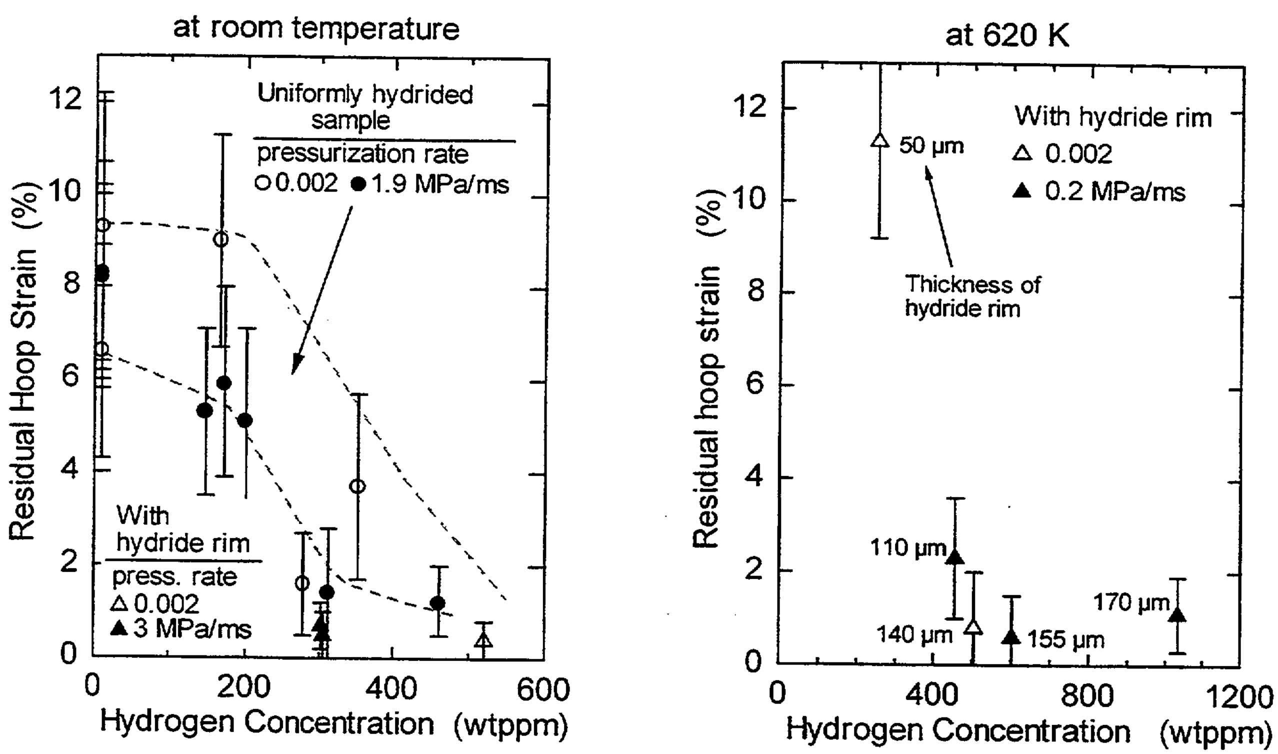


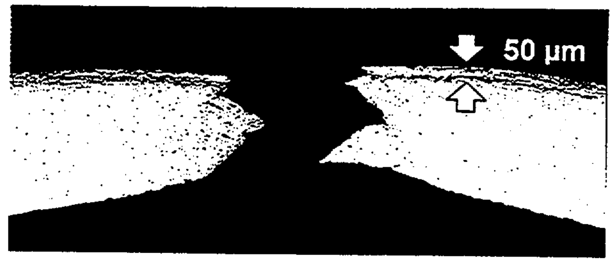
Fig. 12 Transient histories of sample internal pressure measured in the test at 620 K.



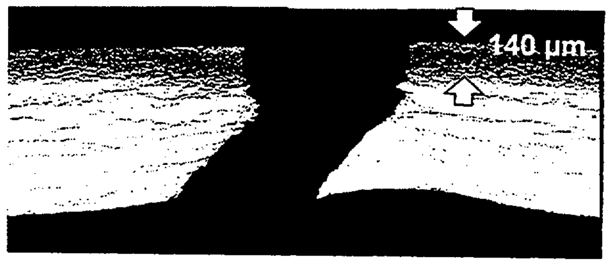
(a) at room temperature

(b) at 620K

Fig. 13 Residual hoop strain of samples in tube burst test



250 wtppm H, Hoop strain = 11.3%



504 wtppm H, Hoop strain = 0.8%

Fig. 14 Radial cross-sections of failure cracks in two claddings with hydride rim.

IV. SUMMARY

In the FK test series of NSRR experiments, 41 to 56 MWd/kgU BWR fuels were subjected to the pulse irradiation. Although fuel enthalpy exceeded 140 cal/g in the Tests FK-3 and -4, fuels did not fail. Due to the pre-test wider gap in BWR fuels, fuel swelling is less significant than that in PWR fuel tests. Fission gas release in BWR fuels during pulse irradiation is strongly influenced by base-irradiation conditions, e.g. linear heat rate. Fuel with the larger fission gas release during base-irradiation gives the larger fission gas release at pulse.

20 MWd/kgHM MOX fuels were also subjected to the pulse irradiation. Fuel enthalpy reached 140 cal/g in the Test ATR-4, and fuel did not fail. PCMI occurs in the lower enthalpy level, and fuel swelling is larger in the ATR/MOX fuels in comparison with the BWR/FK fuels that have a similar pre-test state. Fission gas release is significantly large in the MOX. Large pores in Pu spot, micro-cracks initiated from the spot, grain boundary separation in the vicinity of the spot are observed in the post-test fuels. These phenomena occurred in the Pu spots have a potential to cause the large swelling and the significantly large fission gas release during the transients.

In the tube burst test with artificially hydrided samples, residual hoop strain of failed tube decreases at the higher hydrogen concentration, and becomes less than 1% for samples with hydride rim. Even at an elevated temperature (620 K), residual hoop strain becomes less than 1% in samples with hydrogen concentration above 500 wtppm (with hydride rim thicker than 140 μm). This fact indicates an important role of the hydride rim in the process on PCMI failure of high burnup PWR fuels.

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REFERENCES

(1) Fuketa, T., Mori, Y., Sasajima, H., Ishijima, K. and Fujishiro, T., "Behavior of High Burnup PWR Fuel Under a Simulated RIA Conditions in the NSRR", Proc. CSNI Specialist Mtg. on Transient Behavior of High Burnup Fuel, Cadarache, France, September 12-14, 1995, NEA/CSNI/R(95)22, pp.59-85, (1996).

(2) Fuketa, T., Nagase, F., Ishijima, K. and Fujishiro, T., "NSRR/RIA Experiments with High-Burnup PWR Fuels", Nuclear Safety, Vol.37, No.4, pp.328-342, (1996).

(3) Fuketa, T., Sasajima, H. and Sugiyama, T., "Behavior of High Burnup PWR Fuels with Low-Tin Zircaloy-4 Cladding Under RIA Conditions", Nuclear Technology, submitted.

(4) Sugiyama, T. and Fuketa, T., "Mechanical Energy Generation during High Burnup Fuel Failure under Reactivity Initiated Accident Conditions", J. Nucl. Sci. Technol., to be published.

(5) Nakamura, T., Yoshinaga, M., Takahashi, M., Okonogi, K. and Ishijima, K., "Boiling Water Reactor Fuel Behavior under Reactivity-Initiated-Accident Conditions at Burnup of 41 to 45 GWd/tonne U", Nuclear Technology, to be published.

(6) Nakamura, T., Yoshinaga, Y., Sobajima, M., Ishijima, K. and Fujishiro, T., "Boiling Water Reactor Fuel Behavior at Burnup of 26 GWd/tonne U Under Reactivity-Initiated Accident Conditions", Nuclear Technology, Vol.108, No.1, pp.45-60, (1994).

(7) Sasajima, H., Fuketa, T., Nakamura, T. and Kikuchi K., "MOX Fuel Behavior Under Reactivity Accident Conditions", J. Nucl. Sci. Technol., submitted.

(8) Schmitz, F. and Papin, J., "High burnup effects on fuel behavior under accident conditions: thetest CABRI REP-Na", J. Nucl. Mater., Vol.270, pp.55-64, (1999).

(9) Cazalis, B., "Status of the interpretation of the MOX CABRI REP-Na tests", 2nd CABRI Technical Review Meeting, Cadarache, France, June 28-29, 1999.

(10) Fuketa, T., Nagase, F., Nakamura, T., Uetsuka, H. and Ishijima, K., "NSRR Pulse Irradiation Experiments and Tube Burst Tests", Proc. 26th Water Reactor Safety Information Mtg., Bethesda, Maryland, October 26-28, 1998, NUREG /CP-0166, Vol.3, pp.223-241, (1999).

(11) Nagase, F., Otomo, T. and Uetsuka, H., "High-pressurization-rate Burst Test of Hydrided Zircaloy-4 Fuel Cladding at Room Temperature", JAERI-Research 98-064, (1998) [Text in Japanese].

Further Results and Analysis of MOX Fuel Behaviour Under Reactivity Accident Conditions in CABRI

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Abstract

A first investigation of the MOX fuel behaviour under RIA conditions has been performed through the three MOX fuel tests of the CABRI REP Na programme. From the preliminary analysis of the presently available results, no evidence for local thermal effects resulting from the heterogeneous fuel structure can be derived. The very high clad straining of the 2-cycles rod (8 % maximum value at the pellet edge in REP Na9) is explained and described by combined effect of thermal expansion and intra-granular gases induced swelling mechanism linked to the high energy deposit. On the other hand, the results of all the tests clearly underline an enhanced fission gas contribution with regard to the UO_2 fuel behaviour. A significant increase of fission gas release is found with the MOX fuel compared to UO_2 fuel at similar burn-up ; such effect is explained considering the presence of a high quantity of gases in inter-granular and porosity bubbles associated to the $UPuO_2$ agglomerates behaviour under irradiation and is much increased at high burn-up. As a consequence of a rapid power transient with fuel heat-up and gas overpressure leading to grain boundary separation, a larger amount of gases can be available for clad loading under gas pressure. Moreover, in opposition to the CABRI UO_2 fuel rod failures which occurred when the clad mechanical properties were degraded by the presence of hydride accumulations, the possibility of rod failure with a cladding at a low corrosion level was revealed by the REP Na7 test result with MOX fuel (at 55 GWd/t) with, however, a mean fuel enthalpy at failure time (120 cal/g) higher than the expected maximum enthalpy level in reactor conditions. Consistently with the high fission gas release results, the obtained failure can be explained by a high contribution of the fission gas pressure on the clad loading and suggests a high burn-up effect with MOX fuel.

1- Introduction

The status of the CABRI REP-Na programme, which aims at studying the consequences of reactivity initiated accident in PWRs (RIAs) such as control rod ejection under hot zero power condition, is regularly reported at the WRSM meetings since 1993s.

The main objective of this programme led by IPSN in collaboration with EDF and supported by NRC is the investigation of potential high burn-up effects on fuel behaviour and the verification of the RIA safety criteria for standard UO_2 fuel ([1], [2]). It was launched in 1992 at the request of the French safety authority DSIN, in relation to the intention of Electricité de France (EDF) to increase the discharge burn-up from 47 to 52 GWd/t (mean assembly) and for future fuel managements.

In addition, the investigation of MOX (Mixed Oxide) fuel behaviour under RIA conditions was included in the definition of the REP-Na test matrix, in anticipation of future licensing requests concerning the behaviour of irradiated MOX fuel under RIA conditions.

Indeed, since several years, Pu re-cycling policy led to the use of the MOX fuel in the French PWR plants, according to the "hybrid" fuel management. Nevertheless, in order to get a good energetic equivalence between MOX and UO_2 fuels, the objectives assigned to the MOX fuel are to be equivalent to UO_2 fuel enriched to 3.7% with a fuel burnup increase up to 50 GWd/tM.

This economic aim has to be reached in a complete safety approach, especially for high burnup fuel, considering that MOX fuels present some important differences (fission product accumulation, heterogeneity of the MOX fuel consequent to the manufacturing process) comparatively to the UO_2 fuel, differences which could have an impact on the safety margins in the case of a Reactivity Initiated Accident (RIA).

In the present paper we will focus on the main preliminary outcomes of the three MOX fuels tests performed at different burnup levels (28, 47 and 55 GWd/t local values), as deduced from the first results analysis including available PIE (still underway) and based on some SCANAIR calculations [3].

2- The characteristics of the CABRI REP Na MOX fuel tests

The three MOX fuel experiments performed in the CABRI REP Na programme used MOX fuel rods elaborated through the MIMAS/AUC fabrication process which is characterised by the mechanical mixing of depleted UO_2 with a $(UPu)O_2$ masterblend powder resulting from ball milling of UO_2 (70%) and PuO_2 (30% maximum) powders.

This process leads to a slightly heterogeneous fuel structure with presence of Pu rich particles embedded in the UO_2 matrix. The as-fabricated plutonium content in the clusters is the one of the masterblend, approximately 30%. The mean size of the clusters is about 20µm for the major part, but a maximum of 2% of the clusters may have a mean size higher than 100µm according to the fabrication specifications. The volumic proportion of the agglomerates depends on the Pu content and is about 22% (for 6% average Pu content in the presently tested rods).

Due to the high burnup reached in these clusters (greater than 160 GWd/tM for a 3 cycles rod) high porosity is observed with the appearance of the "rim" zone of a high burnup UO_2 fuel and probably a high fission gas content. Helium production is also significantly higher in MOX fuel rods than in UO_2.

The three REP Na MOX fuel tests used rods irradiated in commercial Nuclear Power Plants at various burnup levels (28, 47 and 55 GWd/t local values) and with standard Zircaloy-4 cladding.

The refabrication procedure in all cases was the FABRICE routine as in all the REP Na tests. In all cases also, the father rod was sectioned at the level of span 5, the penultimate intergrid span at the upper end of the fuel rod. At this level, the burnup is rather constant and the outer clad corrosion is close to rod maximum. The refabricated rodlets were filled with He at 0.3 MPa pressure, equivalent to the sodium pressure in the coolant channel. Last but not the least, all the MOX tests were performed with a power pulse width close to 40 ms, a value which is a correct approach of the reactor pulse when loaded with MOX fuel.

The main test characteristics and results are given in the following table 1.

Table 1: The MOX-fuel tests of the CABRI REP Na test matrix

Test (date)	*Tested rod*	*Pulse width (ms)*	*Energy at pulse end (cal/g)*	*Clad-corrosion (μmZrO2)*	*Results and remarks*
Na-9 (4/97)	EDF MOX 2 cycles span 5 28 GWd/t	34	197 (at 0.5 s) 241 (at 1.2 s) (1007 J/g)	<20	No rupture Hmax = 210 cal/g ΔΦ/Φ = 7.4 % mean max FGR = 34 % (estimation) Examination currently carried out
Na-6 (3/96)	EDF MOX 3 cycles span 5 47 GWd/t	35	126 (at 0.66 s) 165 (at 1.2 s) (690 J/g)	35	No rupture Hmax = 148 cal/g ΔΦ/Φ (max) : 2.5 % mean FGR = 21.6 %
Na-7 (2/97)	EDF MOX 4 cycles span 5 55 GWd/t	40	125 (at 0.48 s) 175 (at 1.2 s) (732 J/g)	50	Rupture at 120 cal/g Hmax = 140 cal/g Strong flow ejection Pressure peaks Fuel dispersal Examination currently carried out

Nota : the burnup indicated in the table is the maximum burnup of the test rodlet : the father rod has a burnup 10 to 12% lower.

3 – The main preliminary results and analysis of the CABRI REP Na tests with MOX fuel

3.1 REP Na9 test

The REP Na 9 test used a 2-cycles MOX fuel rod irradiated in St Laurent B1 EDF power plant. The test rod was reconditioned from the 5th span of an industrial rod with 28GWd/t local burn-up and a low clad corrosion (~10μm ZrO2).

The power transient of 34 ms half width did not lead to rod failure although the high energy injection (241 cal/g at 1.2s) resulted in a maximum mean fuel enthalpy of 210 cal/g (SCANAIR evaluation).

The maximum fuel and clad elongations are 11 mm and the residual clad elongation amounts to 6 mm.

The transient evolution of the inlet and outlet sodium flow rates showed rapid variations ("TOP effect") which are known to result from the transient radial deformation of the rod, the thermal expansion of the sodium and the heating of the outer channel wall.

The profilometry of the rod after test showed a very high plastic circumferential strain (the maximum value ever obtained in the CABRI REP Na tests) with maximum strain near pellet ends [4]. At the peak power level, the hoop strain reached 8% at the pellet edge, and 6% in the pellet middle height, showing an hourglass type pellet deformation with increase of the primary local strains up to 100µm in diameter (see figure 1). Based on the profilometry on eight diameters, no significant ovalization (<30µm) has been evidenced (figure 2).

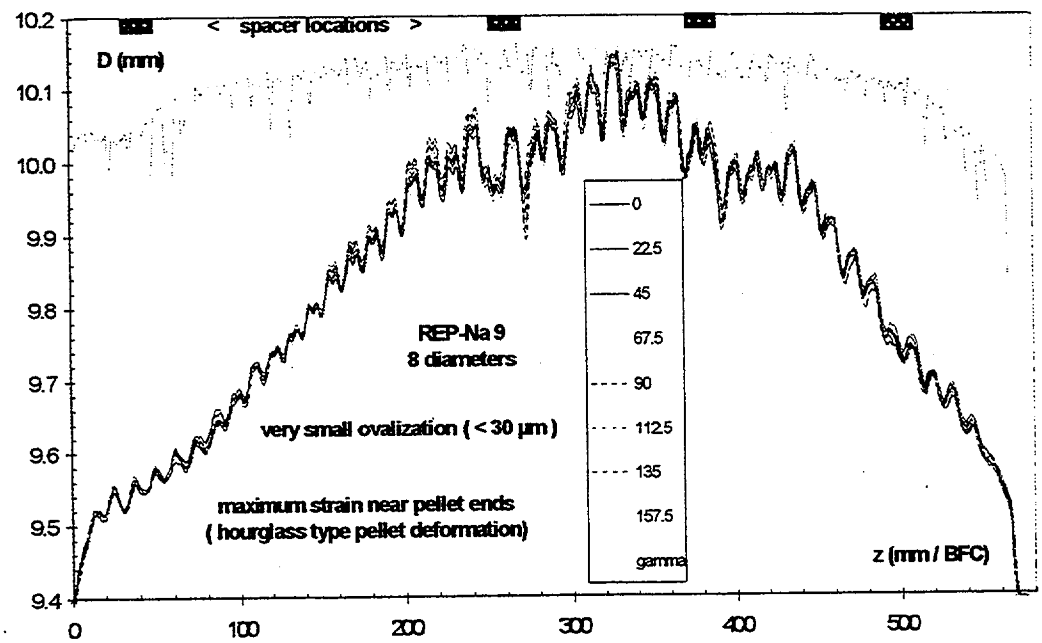

Figure 1 : REP-Na9 profilometry : rod diameter versus axial position

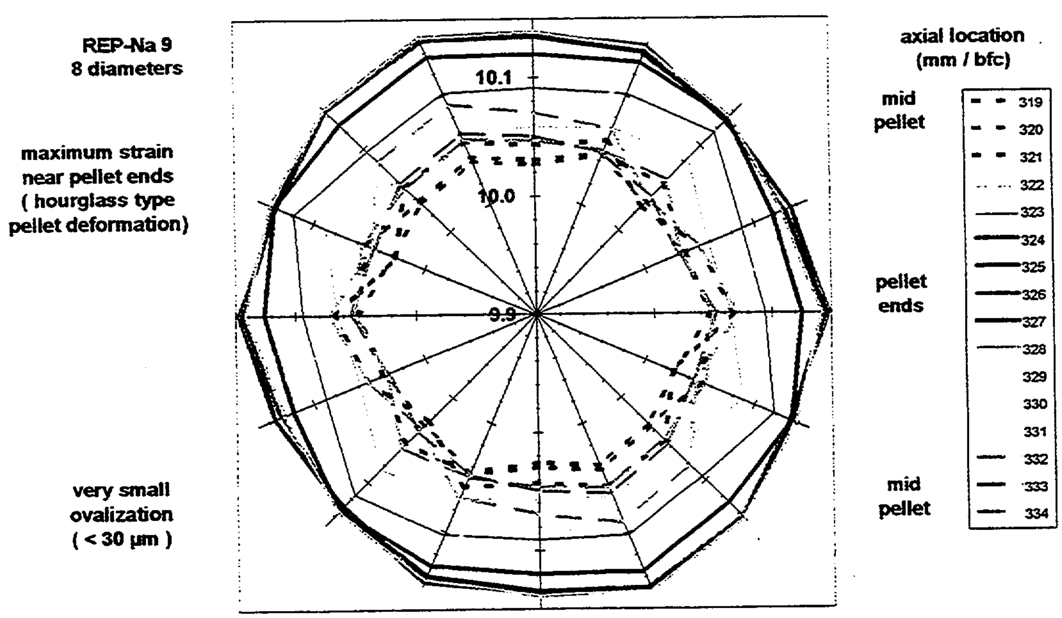

Figure 2 : REP-Na9 profilometry : polar representation of rod diameters versus azimuth

The volumetry of the tested rod indicated a fission gas release (FGR) of about 34% together with a significant Helium release (1/4 of the fission gases).

The preliminary analysis of the destructive examinations showed the following points :

- an almost total filling of the dishings similarly to the REP Na 2 test with UO_2 fuel and similar energy deposit,
- the absence of evolution of the clusters structure indicating no melting occurrence,
- a quasi total grain separation in the UO2 matrix and an important intragranular fission gas precipitation correlated to the high energy level.

The analysis of the REP NA 9 behaviour is to be compared to the REP-Na 2 test performed with an UO2 rod and similar energy deposit : BR3 rod at 33 GWd/t, low corrosion thickness (4 µm), power pulse of 9.5 ms half width with 211 cal/g total energy injected and a maximum mean fuel enthalpy evaluated to be 210 cal/g.

The important energy injection in REP Na9 leads to sustained high temperature levels inside the fuel with maximum fuel temperature of 2700°C in the center part and temperatures higher than 2500°C during more than 1.5 s as shown on figure 3.

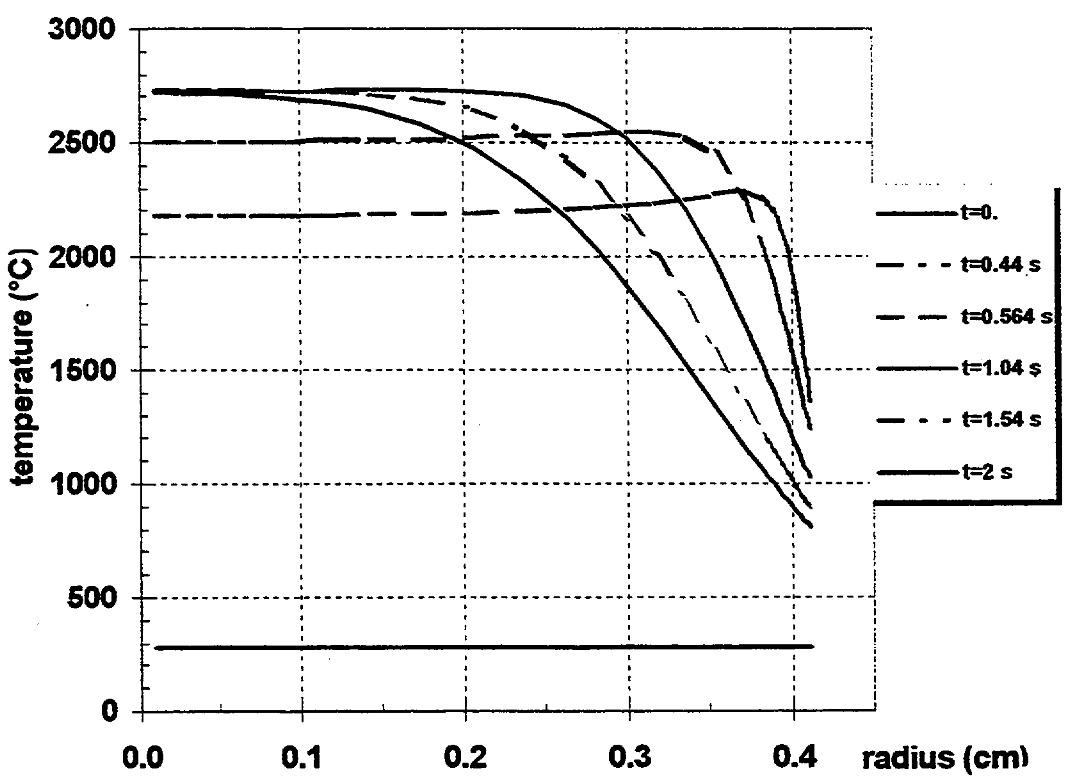

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Figure 3 : REP-Na9 test : radial temperature profiles at PPN at 6 diffe times given by the SCANAIR code.

This induces an important clad straining with hourglass shape similarly to the REP Na2 clad hoop strain, as a result of the fuel loading under parabolic radial temperature profile in the late phase of the transient (fuel cooling down phase).

Such high level of cladding deformation in both tests is explained by the combined effect of the fuel thermal expansion and fission gas induced swelling. This last phenomenon is linked to vacancy diffusion induced by intragranular fission gas bubble pressure increase and is activated when the fuel temperature remains higher than 2100 K during a sufficient time which is the case in the center part of the pellets in both tests with high energy injection (from a SCANAIR evaluation, the contribution of the fission gas induced swelling on the clad deformation accounts for about 70%.)

In spite of the similar energy injection, the higher deformation obtained in REP-Na 9 compared to REP-Na 2 can be explained by the different mechanical properties of the cladding material : indeed, in the temperature range covered by the tests (maximum inner and outer clad temperatures respectively around 900°C and 500°C), the yield stress of the standard zircaloy-4 of REP-Na 9 cladding is strongly reduced in opposition to BR3 rod cladding (by a factor ≈ 2).

Another similar point between the two tests REP Na 2 and REP Na9 is the clear evidence of filling of the dishings consistent with the visco-plastic fuel behaviour at high temperature level, (~ 2700°C).

The most striking difference between the MOX fuel and the UO_2 fuel rods is the fission gas release rate (34% in REP Na 9, 6% in REP Na 2) which is understood as the consequence of the high gas retention inside the porosity of the $UPuO_2$ clusters as evidenced from EPMA examinations of 2 and 3 cycles MOX fuel rods.

This high quantity is thus available to be released in correlation with the grain boundary separation occurrence as already deduced from the analysis of high burn-up UO_2 fuel submitted to a RIA. The total FGR result after the power transient, is thus consistent with the assumption of the contribution of initial porosity and intergranular bubbles associated to grain boundary separation and is increased by intragranular bubbles migration at such high temperature levels as in REP-Na 2 and REP-Na 9.

Finally, its is important to notice that no specific impact of $UPuO_2$ particles with regard to local melting or rod failure has been evidenced in the REP Na 9 in spite of the high energy deposit and of the fuel heterogeneity.

3.2 / REP Na6 test

The REP-Na 6 test has been performed with a 3 cycles MOX fuel rod irradiated in the St Laurent B2 EDF power plant. The test rod reconditioned from the 5^{th} span of a reactor rod was characterized by a burn-up level of 47 GWd/t and a maximum clad corrosion thickness of 35 µm.

The CABRI power transient of 40 ms half width injected 165 cal/g at the peak power level (at 1.2s) resulting in a maximum mean fuel enthalpy of 148 cal/g (SCANAIR evaluation).

The rod did not fail but a significant residual clad straining (figures 4 and 5) has been obtained (2.5% mean value at peak power node, PPN) together with a strong ovalization (100 µm in diameter at PPN).

The origin of such ovalization is not yet understood and studies are underway in order to identify whether this can be due to fuel or clad heterogeneity (microstructure, gas retention, corrosion), or to asymmetrical energy deposit in the CABRI reactor (neutron absorption due to instrumentation devices). The fact that both tests with higher energy deposit and lower burn-up such as REP Na 2 and REP Na 9 do not lead to straining with ovalization is also to be considered (deformation with contribution of the fuel center part, more homogeneous).

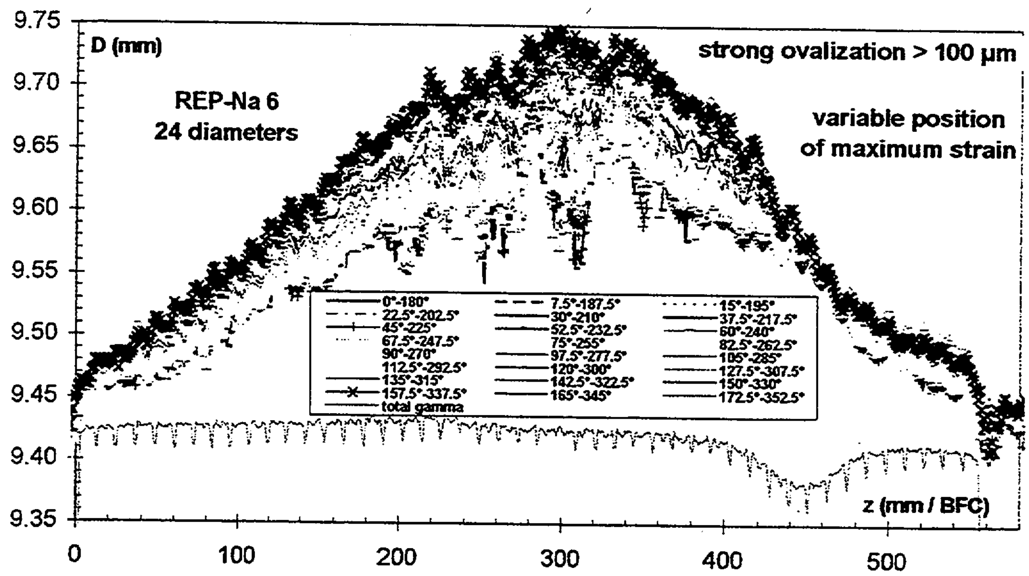

Figure 4 : REP-Na6 profilometry : rod diameter versus axial position

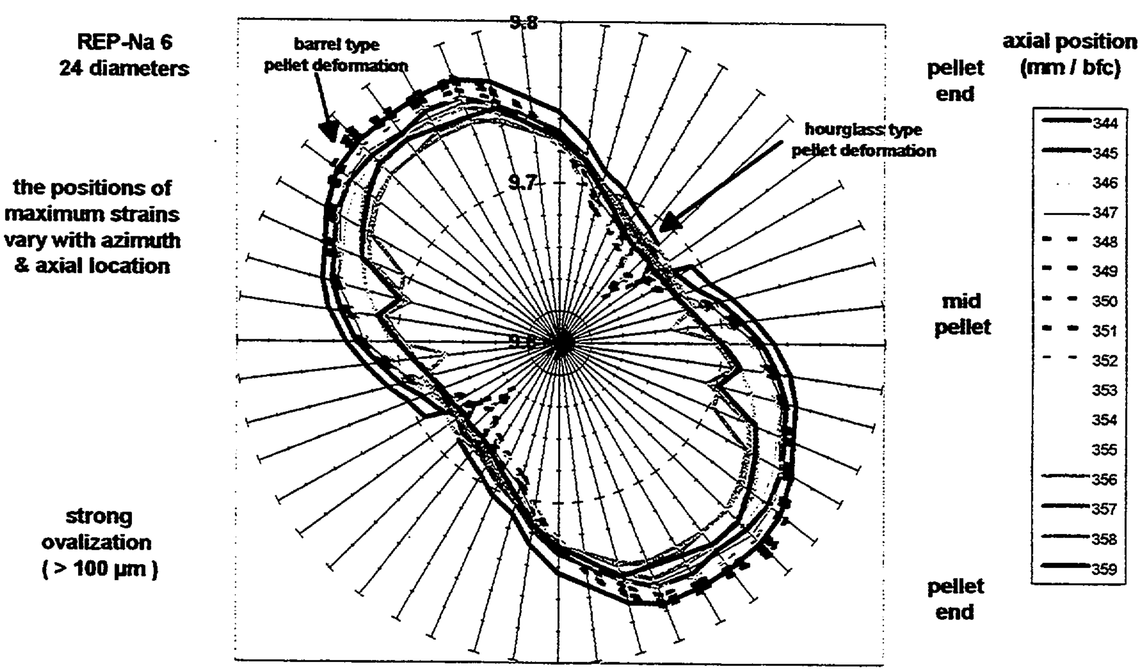

Figure 5 : REP-Na6 profilometry : polar representation of rod diameters versus azimuth

The test was also characterised by a significant oxide spalling close to the peak power node (PPN) and to the top of fissile length as deduced from the zirconia thickness measured after test.

A significant fission gas release was also evidenced (21.6 % of the retained gases).

The post-test metallographic examinations clearly showed a high rate of grain boundary failure in the UO_2 matrix particularly at the pellet periphery linked to bubble overpressure (see figure 6) as previously seen in the other REP Na tests.

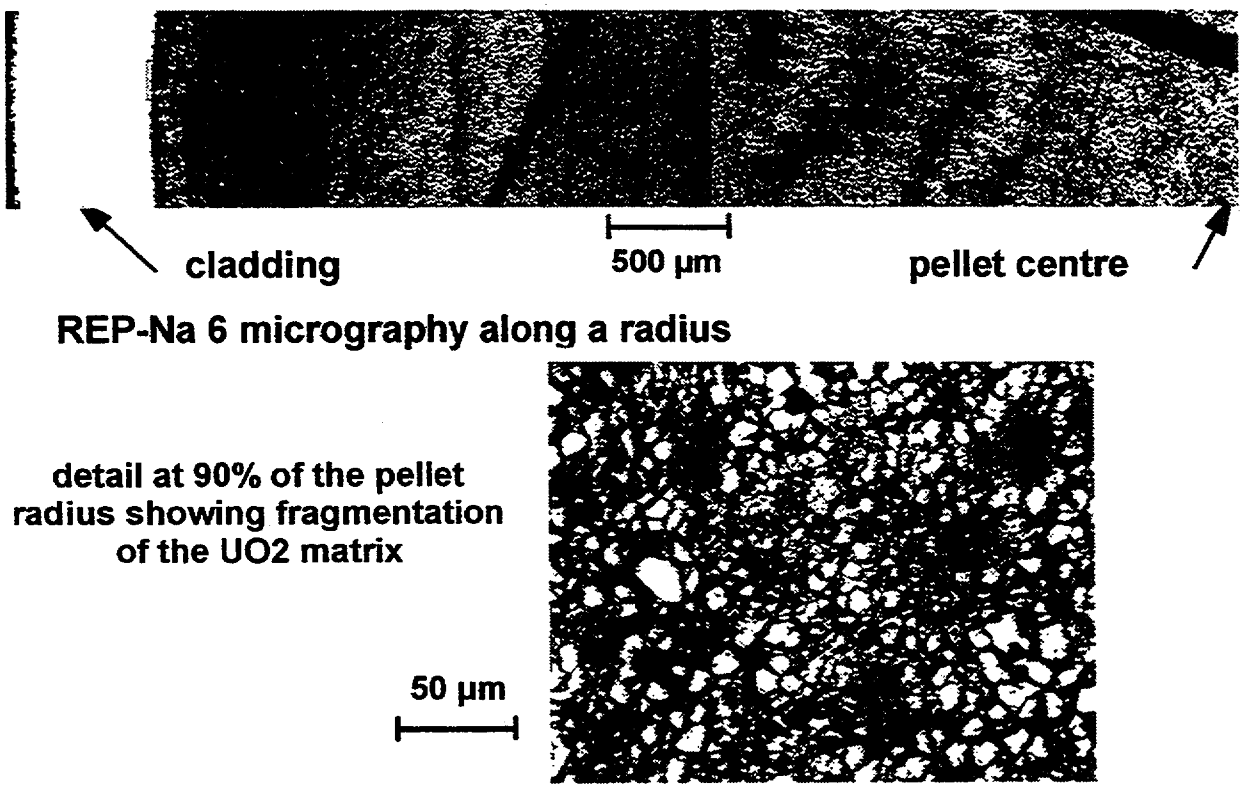

Figure 6 : REP-Na 6 radial cut at peak power location

Another evidence is the inter and intragranular gas precipitation in the UO_2 matrix with presence of channels at grain boundaries (see figures 7a and 7b) which is a clear indication of high temperature level (> 2100 K) and confirms the contribution of the center part to intragranular gas swelling and the global fission gas release.

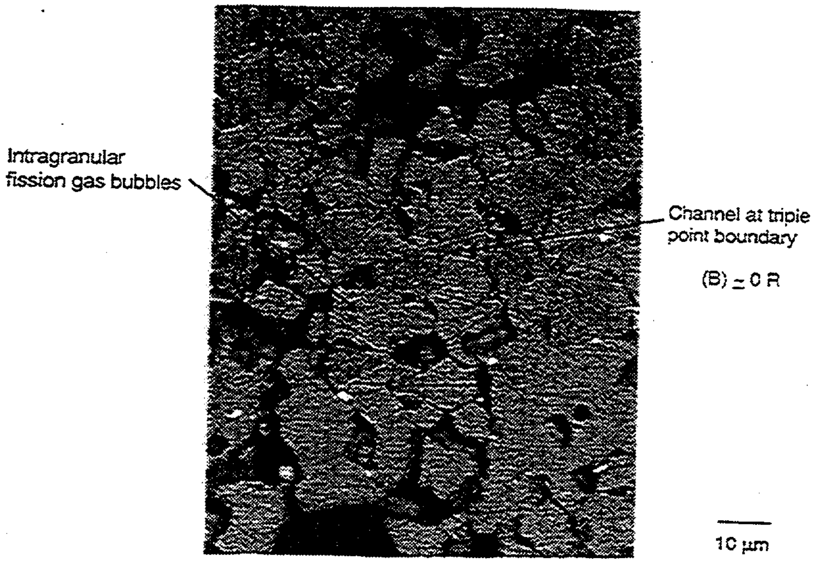

Figure 7a : REP-Na 6 micrograph (central part)

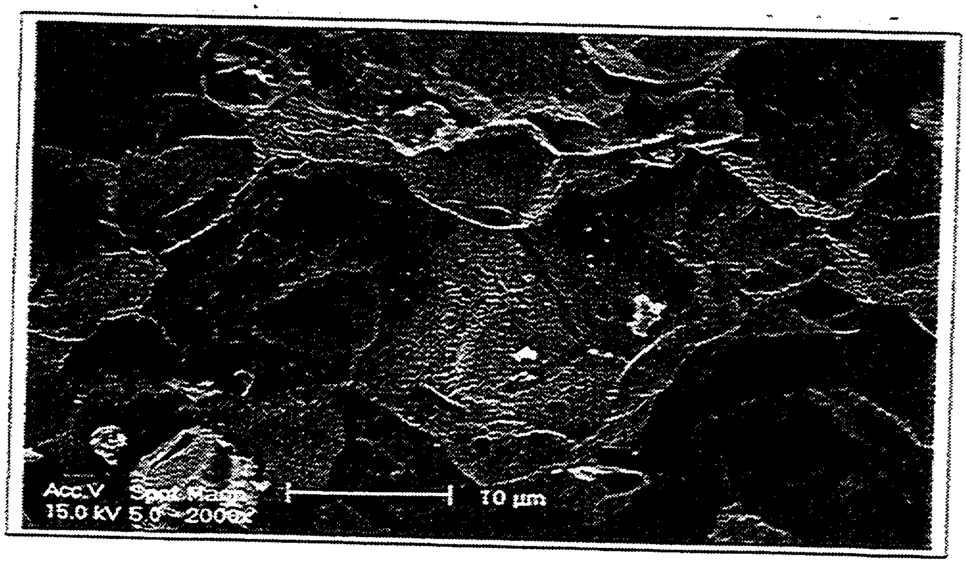

Figure 7b : gas paths on triple boundaries at 4000 µm/edge of the pellet

These results are consistent with the thermal description of the REP Na6 test given by the SCANAIR code which showed that the energy injection resulted in a rapid increase of the fuel temperature in the pellet periphery up to 1700°C and that a maximum fuel temperature of 2100°C is reached in the center part in the late phase of the transient.

The significant amount of FGR is also consistently explained by the important grain boundary separation observed in the test together with the clad straining allowing gas to escape through pathes and by the contribution of the center part fuel (channels at grain boundaries, figure 7).

Finally, the examinations did not highlight any specific thermal behaviour of the Pu aggregates which did not undergo local melting (even when located at the pellet periphery), nor significant morphology modification as compared to the pre-irradiation state.

3.3 / REP-Na 7 test

The REP-Na7 test has been performed using a 4-cycles MOX fuel rod (55 GWd/t local value) reconditioned from the 5th span of a PWR rod irradiated in Gravelines 4 power plant and with a clad corrosion thickness of 50 µm.

The neutron-radiography before the test did not exhibit any hydride accumulation (so called « blister ») nor spalling of the oxide layer.

The power transient of 40 ms half width led to rod failure (at 452 ms) for an injected energy of 109 cal/g at peak power node. According to calculations with the SCANAIR code [3] the rod failed at the time when a mean fuel enthalpy of 120 cal/g at PPN was reached (such value is however higher than the maximum expected enthalpy in reactor conditions).

The failure was immediately followed by a strong sodium flow ejection and high pressure peaks in the channel (200 b at inlet, 110 b at outlet) and by the voiding of the coolant channel [4].

From the microphones and flowmeter signal analysis, the failure has been located around PPN (26 cm from bottom of fissile length). However, the hodoscope did not give any evidence of fuel motion at that time due to its low sensitivity with low enriched PWR fuel.

A second event, in the lower part of the test rod, occurred 18 ms later (seen by hodoscope, flowmeter, pressure transducers), clearly indicating fuel motion in the lower part of the channel : at this time, which could be considered as the latest one for the onset of fuel ejection, the maximum fuel enthalpy is evaluated to be 130 cal/g.

The large amount of fuel motion is confirmed by the low residual sodium flow (5% of its initial value) indicating an almost complete channel blockage. This point is corroborated by the non-destructive examinations showing loss of fuel in the lower part of the fissile column and fuel relocation in the filters.

Due to the delay of hot cell work the post-test examinations of REP Na7 are currently underway.

The main striking point is the rod failure which occurred in spite of a limited corrosion level and absence of spalling of the cladding, in opposition to the UO_2 fuel REP Na failed rods characterized by a high burnup level (60 GWd/t) and a thick corrosion layer (80 – 130 µm) with initial spalling leading to the presence of "blisters" (hydride accumulation).

First of all, in REP Na 7, the axial location of the first failure at peak power level tends to eliminate any effect of fuel heterogeneity due to $UPuO_2$ agglomerates which is stochastic and could trigger the failure at any axial level. Moreover, the comparison of REP Na7 and REP Na 6 with a similar power pulse leading to a maximum fuel enthalpy of 145 cal/g without failure, suggests an important influence of the burnup on the clad loading.

In addition, the comparison of the experimental total clad hoop strain evolution versus energy injection clearly indicates (see figure 8) an earlier and higher clad loading in REP Na7 as compared to REP Na6, on the basis of the sodium volume expansion variation.

This point suggests an enhanced clad loading mechanism in REP Na7 in comparison to REP Na6 behaviour.

On the other hand, the results of the mechanical tensile tests realized in the PROMETRA programme on similar cladding as REP Na7 rod showed values around 1% and 30% for uniform and total elongations respectively in the temperature range of 400-600°C : this allows to deduce that the clad failure in REP Na7 results from the contribution of a strong fission gas pressure on clad loading, most probably induced by the high gas pressure inside the porosities of $UPuO_2$ large clusters and by the fission gas availability due to the important grain boundary separation.

The high level of confinement and concentration of gases could also explain the violent flow ejection.

However, additional information from post-test examinations is needed for better understanding.

The failure of the REP Na7 rod can also be opposed to the REP Na3 test result in which a UO_2 3 cycles rod (53 GWd/t zircaloy low tin cladding and corrosion thickness of 40 µm without spalling) submitted to a power pulse of 10ms half width with 120 cal/g injected at PPN and a maximum fuel enthalpy of 125 cal/g, did not lead to rod failure : such comparison tends also to confirm an enhanced clad loading mechanism in MOX fuel comparatively to the UO_2 fuel behaviour.

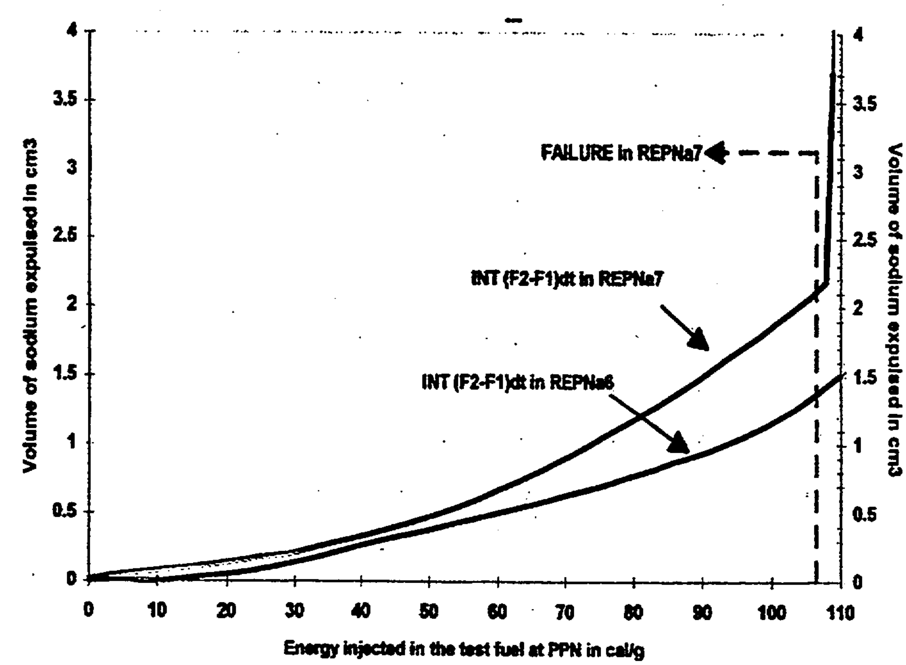

Figure 8 : volume of sodium expulsed because of clad swelling just before the fail

4 / Summary and preliminary outcomes compared to UO_2 fuel behaviour

From the preliminary interpretation of the three CABRI REP Na tests with MOX fuels at various burnup levels, the following points can be deduced.

With regard to thermal aspects, there is no evidence of a direct impact of the $UPuO_2$ agglomerates on rod failure occurrence even when located at the pellet periphery (REP Na6) and in spite of high energy deposit (as in REP Na9).

The occurrence of the transient clad spalling due to clad straining is confirmed as it already occurred with UO_2 fuel rods in REP Na3 and REP Na4. Consequences on the clad-coolant heat transfer in reactor situation might be expected from this behaviour (enhancement of the risk of boiling crisis followed by clad heat-up, debris accumulation).

Similarly to what has been observed with UO_2 fuel CABRI tests, the contribution of intragranular gases to fission gas induced fuel swelling is confirmed at high temperature level (mainly in REP Na2 and REP Na9 and for a less part in REP Na6) leading to significant clad straining, in addition to the fuel thermal expansion on the basis of fuel microstructure after tests.

Fuel viscoplastic behaviour is observed at high temperature level and results in the filling of the dishings, similarly in UO_2 and MOX fuels (REP Na2 – REP Na9) : such effect in reducing the free volume, might lead to transient internal pressure increase.

A significant increase of fission gas release is found with the MOX fuel compared to UO_2 at similar burnup ; such result is understood as the result of the presence of a high quantity of gases in intergranular and porosity bubbles associated to the $UPuO_2$ agglomerates behaviour under irradiation which is much increased at high burnup. As a consequence of a rapid power transient with fuel heat-up and gas overpressure leading to significant grain boundary separation in the UO_2 matrix as seen in REP Na6 and REP Na9, a larger amount of gases can be released and be available for clad loading under gas pressure.

The failure of the REP Na7 test rod with a sound cladding (without hydride blisters) and a low corrosion level may be explained by the contribution of a high gas pressure on clad loading and suggests a high burnup effect with MOX fuel.

5 - Conclusion

The three MOX fuel tests of the CABRI REP Na test programme have allowed to investigate the consequences of the combined thermal and fission gas effects on a MOX fuel rod submitted to a RIA.

From the experimental findings which are presently available, no evidence for thermal effects resulting from the heterogeneous nature of the fuel can be given.

On the other hand, there are very clear hints that fission gas effects are enhanced with regard to the behaviour of UO_2, when comparing the fission gas release rate of the unfailed tests or the failure conditions of REP Na7.

These increased fission gas effects are to be explained by the significant difference of the gas retention in MOX fuel during nominal operation. The heterogeneous nature of the MOX and also the slightly higher fuel temperature during nominal operation produce, for a given gas release fraction, a higher gas retention fraction in intergranular bubbles and in porosities. These gas retention sites produce undelayed pressure effects during rapid transient heating which lead to grain boundary separation, gas availability and associated clad loading mechanism.

The failure of REP Na7 is to be considered with particular attention because it is of a fundamentally different type than the failures of UO_2 fuel rods which only occurred when the clad mechanical Mechanical properties were severely affected by the presence of hydride blisters. In REP Na7, the loading potential due to gas retention inside the intergranular bubbles and porosities was high enough to break a sound cladding. Due to the similarity of structures in the MOX clusters and in the rim zone of a high burnup UO_2 fuel, similar loading mechanism might be expected at very high burnup for UO_2 (>80 GWd/t).

Nevertheless, further studies are needed to better quantify the initial rod states concerning the gas repartition and to confirm the loading mechanism.

In the next year, an additional test with a MOX fuel rod of similar fabrication than the presented rods (MIMAS/AUC) but at a higher burnup level (5 cycles), is planned in the CABRI sodium facility and should allow to complement the data base in the range of burnup levels foreseen for future fuel managements.

In parallel, the support experimental programme planned in the SILENE reactor with capsules tests submitted to fast power transients for the study of fission gas transient behaviour will also help in the understanding and quantification of the basic mechanisms.

Lastly, the future CABRI Water Loop experimental programme under international cooperation should also allow to address the MOX fuel behaviour with high burnup and improved fuel microstructures under representative conditions.

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References

[1] J. PAPIN – F. SCHMITZ
The status of the CABRI REP Na test programme : present understanding and still pending questions
WRSM 25th, October 97, Bethesda, USA

[2] F. SCHMITZ – J. PAPIN
REP Na 10, another RIA test with spalled high burnup rod and with a pulse width of 30ms
WRSM 26th, October 98, Bethesda, USA

[3] J. PAPIN – H. RIGAT – F. LAMARE – B. CAZALIS
The SCANAIR code for the description of the PWR fuel rod behaviour under RIA : validation on experiments and extrapolation to reactor conditions
LWR Fuel Performance, March 97, Portland, USA

[4] F. SCHMITZ - J. PAPIN – Ch. GONNIER
RIA tests in CABRI with MOX fuel
AIEA Symposium on MOX fuel, May 99, Vienna, Austria

SUMMARY OF RESULTS ON THE BEHAVIOR OF VVER HIGH BURNUP FUEL RODS TESTED UNDER WIDE AND NARROW PULSE RIA CONDITIONS

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Abstract

Capsule tests performed with high burnup VVER fuel rods in the impulse graphite reactor (IGR) under reactivity initiated conditions demonstrated the leak of the pellet cladding mechanical interaction (PCMI) failure [1]. High temperature mechanism of the cladding failure due to plastic deformation and rupture was fixed only for this type of fuel rods. The analysis of representativity of these tests showed that:

- all tests were performed under wide pulse conditions;
- specific behavior of gas fission products can lead to an additional stress of cladding during the PCMI stage under narrow pulse conditions.

That is why the new tests with VVER fuel rods were performed in special impulse reactor (BIGR) under narrow pulses. Major parameters of these tests were:

- number of fuel rods – 6;
- burnup – 48, 61 MWd/kg U;
- pulse width – 2.6 - 3.2 ms;
- peak fuel enthalpy – 114 - 153 cal/g.

All six fuel rods did not fail after the tests.

Introduction

During last years two research programs were realized in Russia to study the high burnup fuel behavior under Reactivity Initiated Accidents (RIA). The organization structure of these programs is presented in Fig. 1[1, 2]. Both of these programs were performed with VVER refabricated fuel rods under capsule test conditions (0.1 MPa, 20°C, stagnate water, a power pulse of the IGR or BIGR research reactors). Results of the first of programs allowed to formulate the following conclusions:

- the mechanism of VVER high burnup fuel rod failure is a high-temperature rupture of cladding (HTRC);
- the peak fuel enthalpy of the failure is about 160 cal/g for pressurized fuel rods.

Thus, these tests have demonstrated that the behavior of VVER high burnup fuel rods (50 MWd/kg U) is similar to the behavior of VVER unirradiated fuel rods (see Fig. 2). However, the comparative analysis of the IGR test results and corresponding results obtained from the tests with PWR high burnup fuel rods in the CABRI [3] and NSRR [4] reactors showed that:

- different mechanisms of failure were observed (pellet cladding mechanical interaction (PCMI) for PWR fuel rods and HTRC for VVER fuel rods);

- different failure thresholds were determined (30–80 cal/g for PWR fuel rods and 160 cal/g for VVER fuel rods).

1990–1996 → Program of tests in the IGR reactor

↓

1. Russian Research Center “Kurchatov Institute” (Moscow)
2. Institute of Atomic Reactors (Dimitrovgrad)
3. Institute of Atomic Energy of National Nuclear Center of Kazakhstan Republic (Semipalatinsk)

1997–1999 → Program of tests in the BIGR reactor

↓

1. All-Russian Scientific and Research Institute of Inorganic Materials (Moscow)
2. Russian Research Center “Kurchatov Institute” (Moscow)
3. State Research Center “Research Institute of Atomic Reactors” (Dimitrovgrad)
4. State Research Nuclear Center “Institute of Experimental Physics” (Sarov)

Fig. 1. Structure of Russian reactor tests with high burnup fuel of VVER type.

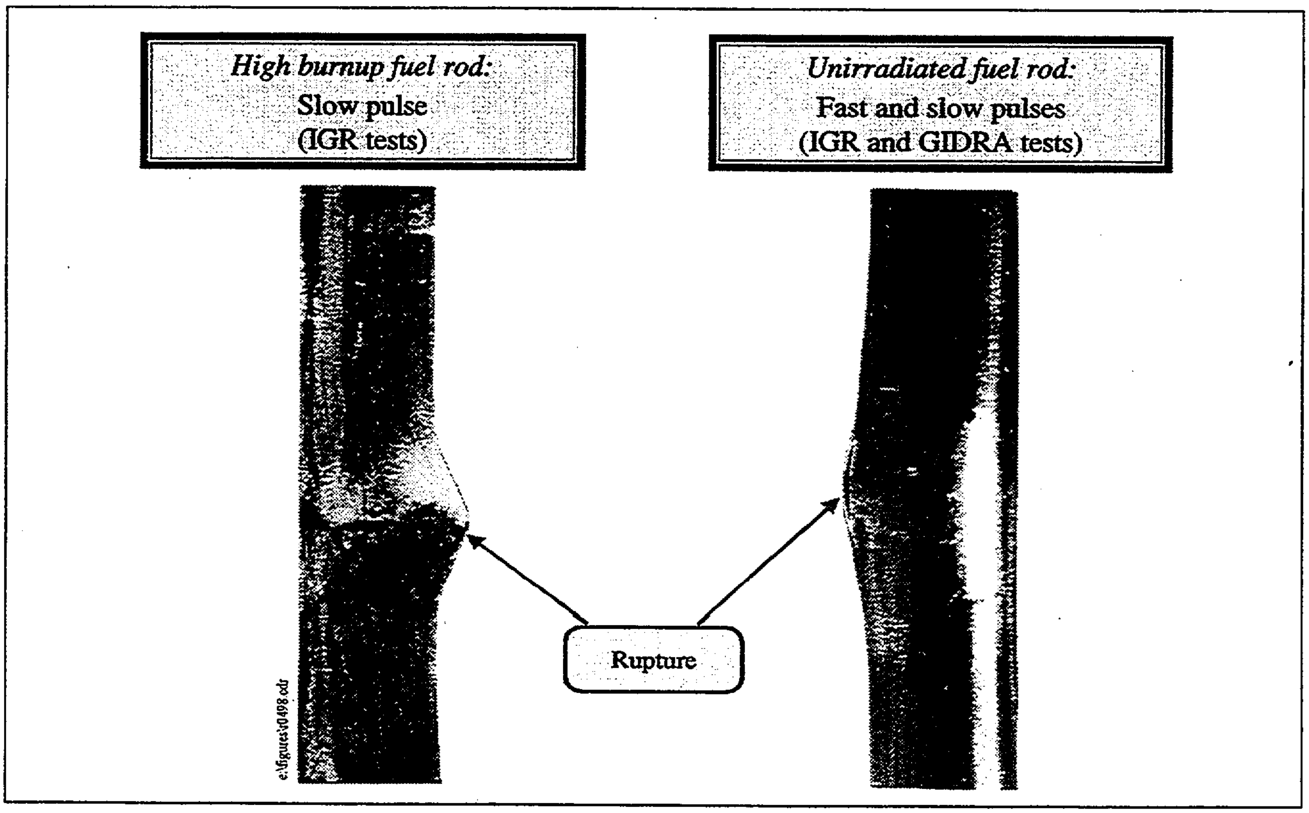


Fig. 2. Failure mechanisms for unirradiated and high burnup fuel rods of VVER type.

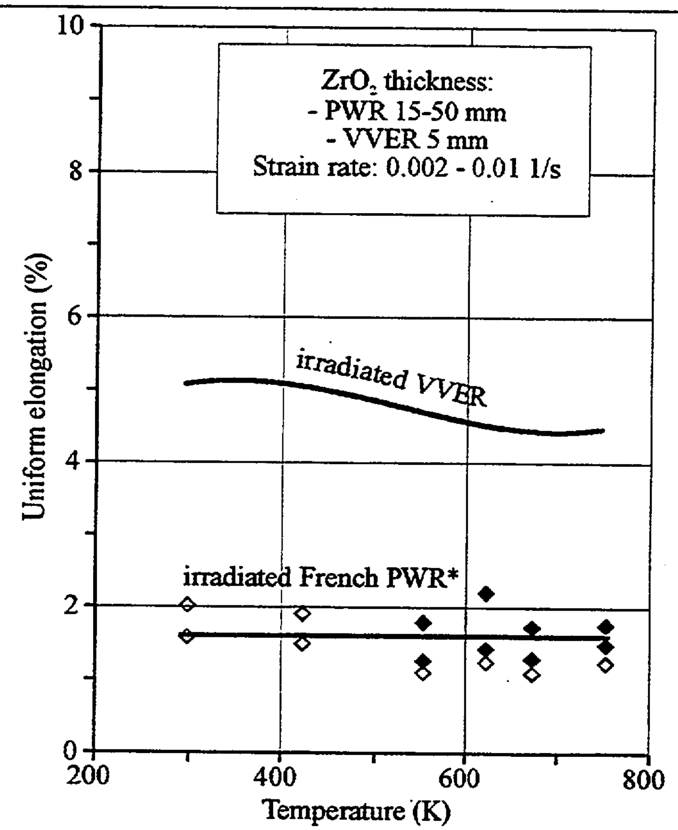
That is why a considerable international discussion was expanded to understand and to explain the RIA test results with VVER and PWR high burnup fuel rods. This discussion, that was accompanied by a series of analytical researches resulted in the conclusion that the following two key factors must be taken into account for the comparative interpretation of VVER/IGR test results [5]:

1. High ductility of Zr-1%Nb irradiated cladding due to the low initial oxidation and hydriding;
2. The slow pulse of the IGR reactor.

The first position characterizes the differences of mechanical properties of Zr-1%Nb and Zry irradiated claddings. Special cycles of measurements performed in Russia [1] and in France [6] allowed to compare the ductility of VVER and PWR claddings. Some results of these investigations are presented in Fig. 3.

These data demonstrate that the VVER irradiated cladding have a large margin of ductility in comparison with the PWR irradiated cladding even for the case when oxidation of PWR cladding is not so high.

However, the second position of the international discussion conclusions required to pay a special attention to the differences between pulse widths for the IGR tests (630–850 ms) and CABRI, NSRR tests (2–80 ms). The demonstration of the problem can be illustrated using Fig. 4-Fig. 6.



* *PROMETRA test results*

Fig. 3. Uniform elongations of the VVER and PWR irradiated cladding vs. temperature [5].

Fuel microstructure and radial distribution of energy deposition in one of high burnup fuel rods tested in the IGR reactors are presented in Fig. 4. As can be observed the maximum of energy deposition is localized in peripheral zone of fuel named "rim zone". This specific effect is determine by a high concentration of Pu isotopes in the rim zone and additional number of fissions on these isotopes.

This effect results in the atypical radial distribution of fuel temperature for the fast pulse (see Fig. 5).

Other words, the fast pulse causes a sharp rise of fuel temperature in rim zone. Besides, it leads to the overpressurization of fuel due to the disbalance between the rate of fission gases generation and the rate of fission gases migration into the gas plenum of fuel rods. Finally, these processes lead to the increase of cladding stresses at the PCMI stage for fast pulses.

To estimate the scale of these effects special calculations using the SCANAIR code [8] were performed. The results of calculations are presented in Fig. 6.

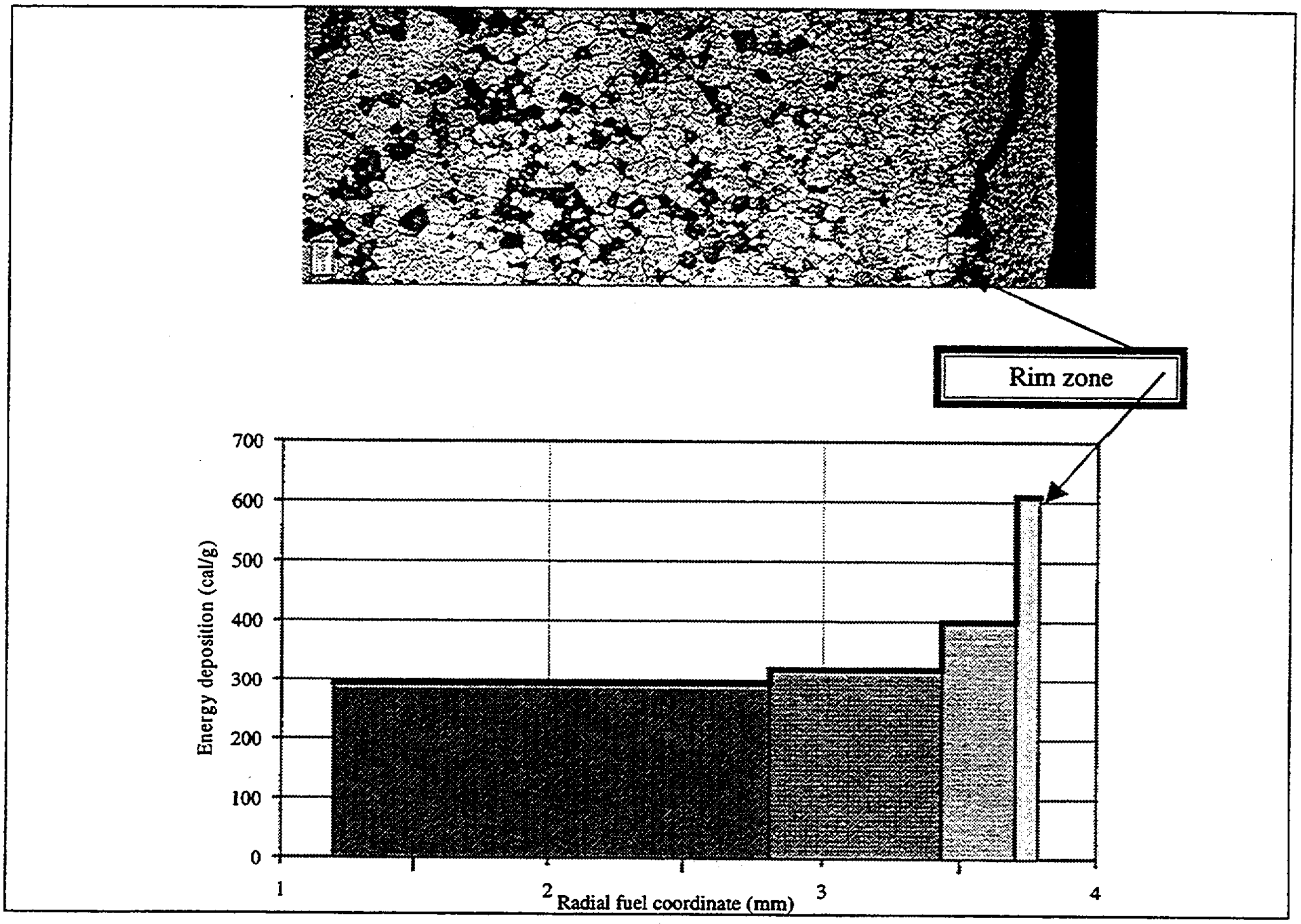


Fig. 4. Fuel microstructure and radial distribution of energy deposition in VVER high burnup fuel rod (50 MWd/kg U) tested in the IGR reactor.

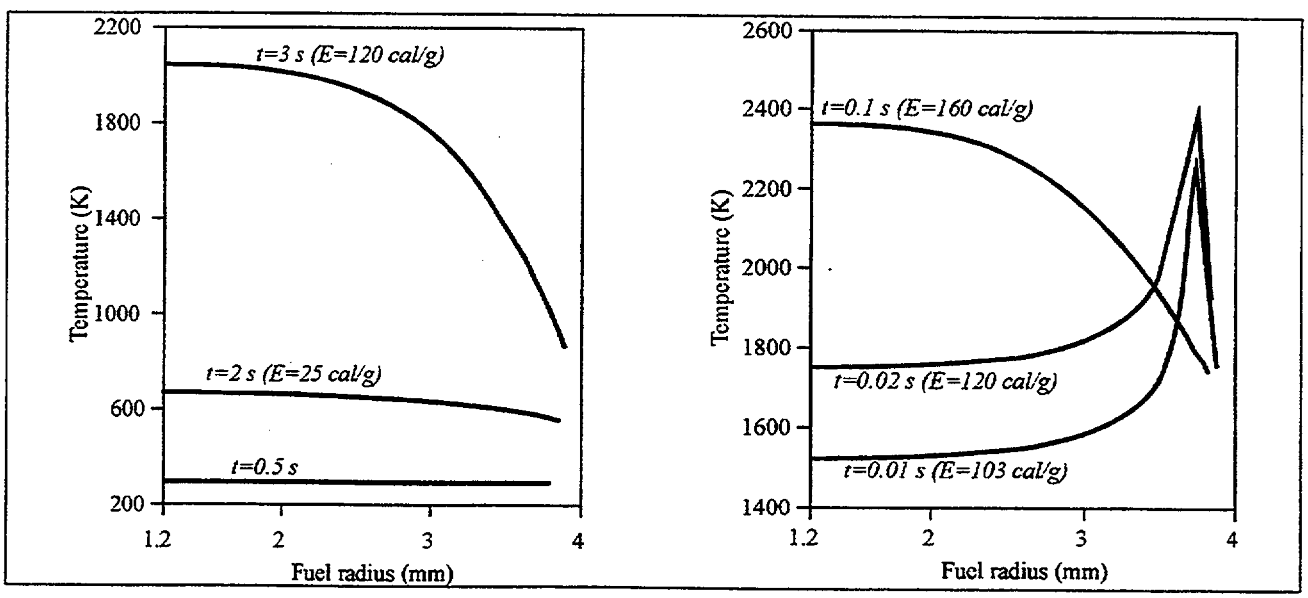


Fig. 5. Distributions of fuel temperature vs. time for slow and fast pulses (peak fuel enthalpy is 160 cal/g).

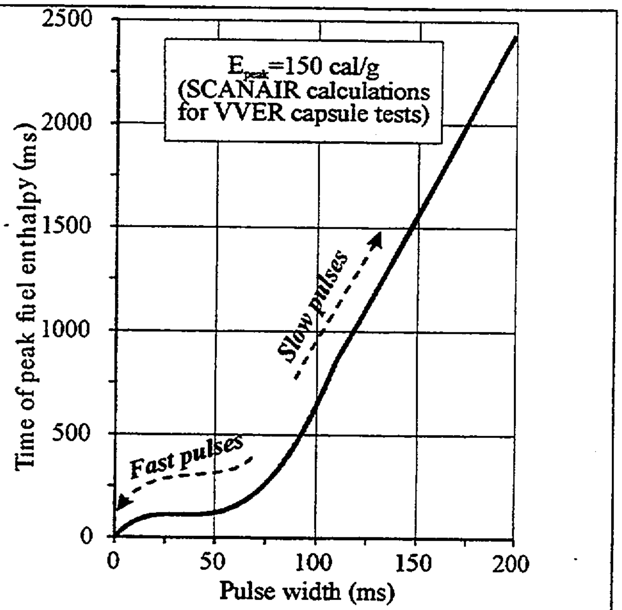


Fig. 6. Time of peak fuel enthalpy vs. pulse width.

This figure demonstrates that phenomena described above can be noted for pulse widths up to 60 ms approximately. Results of these analyses have shown that two possible ways can be selected to validate the behavior of VVER high burnup fuel rods under RIA conditions:

1. To prove that pulse widths are considerably longer than 60 ms for NPP of VVER types.
2. To perform a set of additional reactor tests with VVER high burnup fuel under fast pulse conditions.

Taking into account the whole collection of uncertainties associated with the item 1 the decision was made to carry out tests in the BIGR reactor following the requirements of the item 2.

Table 1 offers the parameters of IGR and BIGR tests.

Table 1. Main parameters of tests in the IGR and BIGR reactors.

Test reactor	Test conditions	Number of fuel rods	Burnup of fuel, MWd/kg U	Pulse width, ms
IGR	<u>Capsule tests with single fuel rods:</u> • water; • ambient temperature; • atmospheric pressure; • no flow. <u>Fuel rods:</u> • refabricated from commercial VVER fuel elements (NV NPP); • internal pressure 1.6–1.9 MPa; • active length 150 mm. <u>Cladding:</u> • ZrO_2 thickness 5 μm.	H1T	51	800
		H2T	50	760
		H3T	50	820
		H4T	50	760
		H5T	50	840
		H6T	50	800
		H7T	47	630
		H8T	48	850
BIGR	<u>Capsule tests with single fuel rods:</u> • water; • ambient temperature; • atmospheric pressure; • no flow. <u>Fuel rods:</u> • refabricated from commercial VVER fuel elements (NV NPP); • internal pressure 2.0–2.5 MPa; • active length 150 mm. <u>Cladding:</u> • ZrO_2 thickness 5 μm.	RT №1	49	2.6
		RT №2	48	3.2
		RT №3	48	2.6
		RT №4	61	2.6
		RT №5	49	2.6
		RT №6	48	2.6

The data presented in Table 1 show that in general only one parameter was varied in IGR and BIGR tests. This parameter was the pulse width:

- 630–850 ms for IGR tests;
- 2.6–3.2 ms for BIGR tests.

In addition, to estimate the reliability margin of VVER high burnup fuel, burnup of one of six fuel rods tested in the BIGR reactor was 61 MWd/kg U. Summarized results characterize the failure threshold of VVER high burnup fuel obtained from IGR and BIGR tests are presented in Fig. 7.

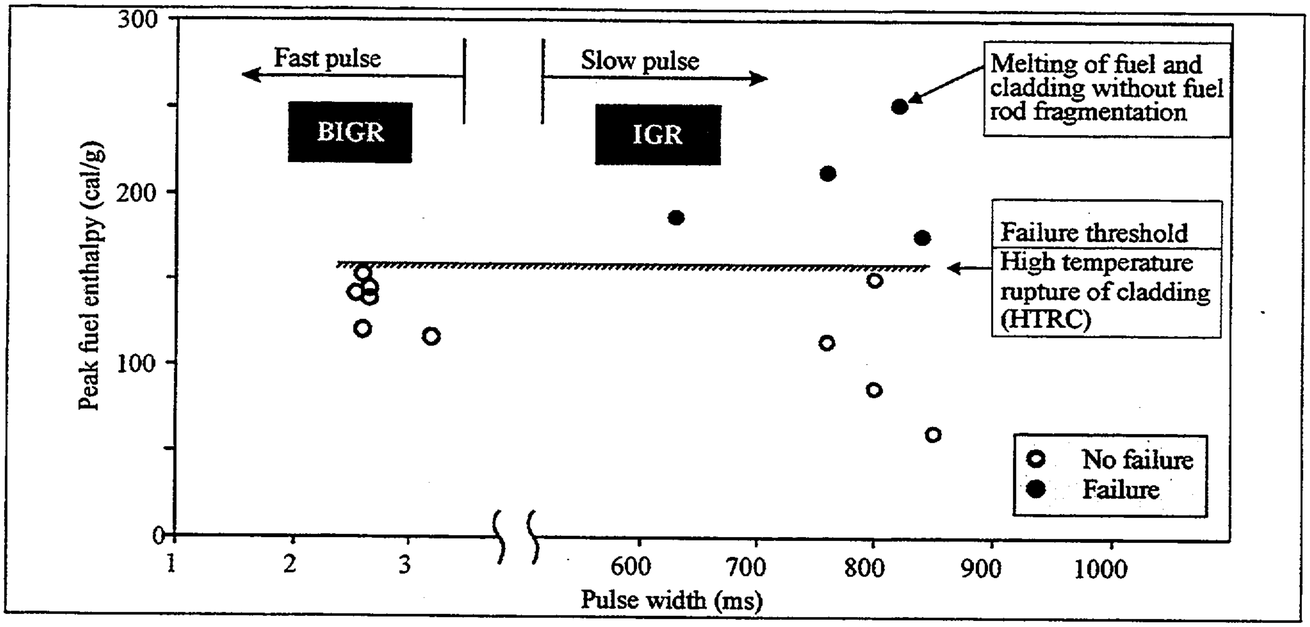


Fig. 7. IGR/BIGR results characterizing high burnup VVER fuel rod behavior under RIA conditions.

As can be seen from these results the PCMI type of failure was not fixed for VVER high burnup fuel rods. Thus, BIGR tests demonstrated clearly that high ductility of irradiated cladding provides for the serviceability of pressurized high burnup fuel rods irradiated up to 50 MWd/kg U and up to peak fuel enthalpy 160 cal/g independently of a pulse width. It is clear that the failure threshold for unpressurized fuel rods will be considerably higher. Also, take into account that the expected failure mechanism for this case is a fuel rod fragmentation due to fuel/cladding melting, it is very important to pay attention on the result of the IGR test with peak fuel enthalpy ~250 cal/g. Post-test examinations of this fuel rod showed that:

- the melting of fuel and cladding was achieved;
- the fragmentation of fuel rod was not achieved.

One more important effect is desirable to be pointed out with the help of data presented in Fig. 8. These data characterize the cladding residual hoop strain versus the peak fuel enthalpy, pulse width, and burnup.

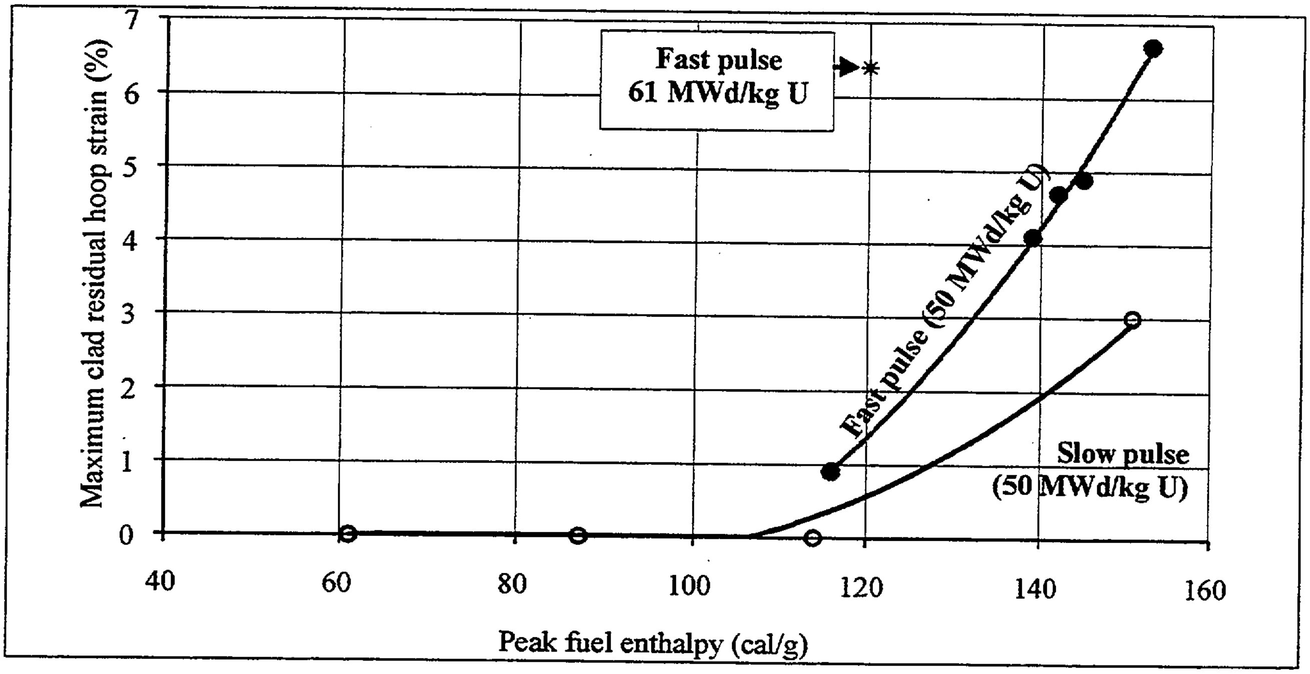


Fig. 8. Zr-1%Nb clad residual hoop strain versus peak fuel enthalpy, pulse width, burnup[1, 2].

The comparison of the data shows a strong dependence between the pulse width and maximum cladding residual hoop strain (a fast pulse leads to the increase in clad strain) and between burnup and cladding strain (the increase in burnup from 50 to 61 MWd/kg U leads to the increase in the cladding strain from 1.3 % to 6.3 %).

Preliminary conclusions and tasks for the future activity

1. BIGR tests have confirmed that high burnup fuel rods with low oxidized Zr-1%Nb claddings tolerate the PCMI stage of loading without a failure
2. IGR and BIGR tests have demonstrated that the failure threshold of pressurized high burnup fuel rods irradiated up to 50 MWd/kg U is about 160 cal/g. The mechanism of failure is a high temperature rupture of cladding.
3. One IGR test has shown that there is no fixed high burnup fuel rod fragmentation at the peak fuel enthalpy of about 250 cal/g. This result and results of out-of-pile mechanical tests showing no differences between mechanical properties of irradiated and unirradiated cladding at high temperatures allow to assume that the threshold of fragmentation due to high temperatures determined earlier for unirradiated fuel rods will be applicable for irradiated fuel rods.
4. Additional efforts should be applied to validate the failure threshold of fuel rods irradiated up to 60 MWd/kg U and higher.
5. The computer analysis of fuel rod behavior under the representative RIA condition should be made as soon as possible.

References

1. L.Yegorova, V.Asmolov, G.Abyshov, V.Malofeev, A.Avvakumov, E.Kaplar, K.Lioutov, A.Shestopalov, A.Bortash, L.Maiorov, K.Mikitiouk, V.Polvanov, V.Smirnov, A.Goryachev, V.Prokhorov, and A.Vurim "Data Base on the Behaviour of High Burnup Fuel Rods with Zr-1%Nb Cladding and UO_2 Fuel (VVER Type) under Reactivity Accident Conditions", NUREG/IA-0156 (IPSN99/08, NSI/RRC KI 2179), Vol. 1-3, 1999.

2. L.Yegorova, E.Kaplar, G.Abyshov, N.Sokolov, O.Nechaeva, A.Salatov, V.Smirnov, A.Goryachev, V.Ustinenko, I.Smirnov, V.Sashnov "Results of tests of VVER high burnup fuel rods in the BIGR reactor under RIA conditions", RRC KI report, 1999 (to be published).

3. J.Papin, M.Balourdet, F.Lemoine, F.Lamare, J.Frizonnet, and F.Schmitz "French Studies on High-Burnup Fuel Transient Behaviour under RIA Conditions", Nuclear Safety, Vol.37, No.4, 1996.

4. T.Fuketa, F.Nagase, K.Ishijima, and T.Fujishiro "NSRR/RIA Experiments with High-Burnup PWR Fuels", Nuclear Safety, Vol.37, No.4, 1996.

5. R.O.Meyer, R.K.McCardell, H.M.Chung, D.J.Diamond, H.H.Scott "A regulatory assessment of test data for reactivity initiated accidents", Nuclear Safety, Vol.37, No.4, 1996.

6. M.Balourdet, C.Bernaudat, V.Basini, N.Hourdequin "The PROMETRA Programme: Assessment of Mechanical Properties of Zircaloy 4 Cladding during an RIA", *SMIRT 15th, Seoul, August 1999.*

7. L.Yegorova, F.Schmitz, J.Papin "Mechanical behavior of fuel element during RIA transients", Proc. of the Eurosafe 1999 meeting, Paris, 1999.

8. K.Mikitiouk, A.Shestopalov, K.Lioutov, L.Yegorova, G.Abyshov "Modification of IPSN's SCANAIR Fuel Rod Transient Code for High Burnup VVER Fuel", NUREG/IA-0165 (RRC KI 2181, IPSN 99/09), 1999.

INDUSTRY STRATEGY AND ASSESSMENT OF EXISTING RIA DATA

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ABSTRACT

A summary of the US nuclear industry strategy to resolve the technical and licensing issues associated with Reactivity Initiated Accident (RIA) events in LWR's is provided. This strategy uses a coupled probabilistic and deterministic approach. Probabilistic Safety Assessments (PSA) are currently under development and will be used to define and rank the risks accompanying SAR Chapter 15 RIA events. The PSA results will provide information to focus research and development activities and set safety criteria. The deterministic approach includes three major components: 1) understand transient fuel rod behavior based on well-characterized RIA-simulation tests, 2) obtain cladding mechanical property data and fission gas release kinetics from separate effects tests, and 3) benchmark the transient fuel behavior code FALCON and perform analytical evaluations of RIA-simulation tests. This comprehensive approach provides a fundamental understanding of the key mechanisms operative during a RIA event.

INTRODUCTION

The goal to achieve higher fuel rod burnup levels has produced considerable interest in the transient response of high burnup fuel. The database on transient fuel behavior is limited at burnup levels beyond 40 GWd/tU and is based on older fuel rod designs. Several experimental programs are underway to generate data on the behavior of high burnup fuel under transient conditions representative of Loss-of-Coolant Accidents (LOCA's) and Reactivity Initiated Accidents (RIA's)[1,2,3]. Such programs include the RIA simulation experiments performed at the CABRI facility in France and the Nuclear Safety Research Reactor (NSRR) in Japan. It is envisioned that the results from these programs will be used to validate analytical codes for high burnup fuel behavior and to develop licensing criteria at extended burnup levels.

Shortly following the first publication of the early experimental results[4,5], the nuclear industry conducted a extensive review and assessment of the observed behavior of high burnup fuel under RIA conditions[6,7,8]. The industry assessment included a review of the fuel segments used in the tests, the test procedures, in-pile instrumentation measurements, post-test examination results, and a detailed analytical evaluation of several key RIA-simulation tests using the EPRI-sponsored transient fuel behavior code FREY. The industry review concluded that loss of cladding ductility due to localized hydrides was the primary cause of failure for high burnup test rods during the RIA-simulation tests[6].

Since the publication of that review, several additional RIA-simulation experiments on test rod segments with burnup levels ranging from 45-64 GWD/T have been performed in the CABRI and NSRR test facilities[9,10]. This paper outlines the industry strategy, summarizes the assessment of the new data, and provides additional interpretations on the response of high burnup test rods during rapid energy deposition events representing a RIA. The relevance of establishing RIA fuel licensing criteria from the results of these RIA-simulation experiments will also be discussed.

Industry Strategy

RIA-simulation tests conducted in France and Japan in the early 90's have raised concerns about the adequacy of the RIA licensing criteria for high burnup fuel[2-5]. The industry strategy to resolve the RIA licensing issues utilizes a coupled probabilistic and deterministic approach. A probabilistic safety assessment (PSA) will be used to define the probability of occurrence and the consequences associated with each Safety Analysis Report Chapter 15 RIA events. By using the event probability and consequences, the risk for a specific event can be defined.and used to rank those events with the highest risk. Such an assessment will depend on the plant type, initiating conditions for an event, and the specifics of the radiological release paths from the plant. In general, the consequences for a RIA range from cladding failure to loss of coolable geometry. The outcome of the PSA will be a risk-informed approach to 1) focus R&D efforts and develop appropriate test conditions, 2) determine the amount of conservatism used in assumptions and analyses, and 3) develop appropriate safety criteria for RIA events.

The Industry effort to assess RIA fuel behavior was initiated in early 1994 following the first tests on high burnup fuel segments[6,7,8]. The deterministic approach employed by the Industry combines elements of experimental results and analytical evaluations to develop a fundamental understanding of fuel behavior during RIA events. The approach includes three major components: 1) use well-characterized RIA simulation tests to establish the transient behavior of the fuel and cladding, 2) obtain data from separate effects tests to define the cladding mechanical properties and transient fission gas release kinetics, and 3) perform benchmarking and analytical evaluations using FALCON. This comprehensive approach provides a mechanistic basis for understanding the key mechanisms operative in RIA-simulation tests and offers a tool to translate experimental results to LWR conditions and different materials, if pertinent material property data are available.

DATA ASSESSMENT

Since the mid-1980's, a total of forty RIA simulation tests have been conducted on LWR-type test rods in the burnup range between 26-65 GWd/tU, including ten tests on BWR-type and three tests on PWR MOX-type fuel. These tests were performed in CABRI with sodium coolant (280°C and 0.5 MPa) or in NSRR in stagnant water (25°C and 0.1 MPa) conditions. All of the test rods were refabricated before testing from full-length fuel rods extracted from fuel assemblies that had been irradiated in commercial PWR or BWR power plants. A majority of these tests are summarized in Tables 1 through 3. Not included in this list is the NSRR JM test series performed on test rods pre-irradiated in the Japanese Materials Test Reactor (JMTR). The JMTR test rods were not included because the irradiation environment was inconsistent with LWR conditions. Also, these tests have been reviewed previously[7,8].

Table 1. CABRI REP Na Test Rods

Test	Peak Pellet Burnup (GWd/tU)	Oxide Layer (microns)	Fuel Type	Pulse Width (msec)	Max. ΔH (cal/gm)	ΔH at Failure (cal/gm)	Fuel Dispersal
Na-1*	65	>80 – spall	17x17	9.5	100	15	Yes
Na-2*	33	10	17x17	9.5	200		
Na-3*	52	50	17x17	9.5	115		
Na-4*	62	85	17x17	70	72		
Na-5*	64	20	17x17	9.1	97		
Na-6	47	35	17x17 – Mox	40	129		
Na-7	55	50	17x17 – Mox	40	140	103	Yes
Na-8*	60	120 - spall	17x17	78	86	57	No
Na-9	28	10	17x17 – Mox	40	183		
Na-10*	64	>80 - spall	17x17	31	94	62	No

*-Analytical evaluations performed on these tests

Table 2. NSRR PWR Test Rods

Test	Burnup (GWd/tU)	Oxide Layer (microns)	Fuel Type	Pulse Width (msec)	Max. ΔH (cal/gm)	ΔH at Failure (cal/gm)	Fuel Dispersal
HBO-1*	50	40-50	17x17	4.4	73	60	Yes
HBO-2*	50	30-40	17x17	6.9	37		
HBO-3*	50	22	17x17	4.4	74		
HBO-4*	50	18	17x17	5.4	50		
HBO-5*	44	35-60	17x17	4.4	80	77	Yes
HBO-6*	49	20-30	17x17	4.4	88		
HBO-7	49	30-50	17x17	4.4	88		
MH-1	39	5	14x14	5.3	47		
MH-2	39	5	14x14	5.0	55		
MH-3	39	5	14x14	4.8	67		
GK-1	42	10	14x14	4.8	93		
GK-2	42	10	14x14	4.8	90		
OI-1	39	N/A	17x17	4.4	106		
OI-2	39	N/A	17x17	4.4	108		
TK-1*	38	7	17x17	4.4	125		
TK-2	48	15-35	17x17	4.4	107	60	Yes
TK-3	50	8	17x17	4.4	99		
TK-4	50	20	17x17	4.4	98		
TK-5	48	25	17x17	4.4	101		
TK-6	38	15	17x17	4.4	125		

*-Analytical evaluations performed on these tests

Table 3. NSRR BWR Test Rods

Test	Burnup (GWd/tU)	Oxide Layer (microns)	Fuel Type	Pulse Width (msec)	Max. ΔH (cal/gm)
TS-1	26	6	7x7	6.7	55
TS-2	26	6	7x7	6.2	66
TS-3	26	6	7x7	5.6	88
TS-4	26	6	7x7	5.0	84
TS-5	26	6	7x7	4.5	98
FK-1*	45	20-40	8x8BJ	4.4	130
FK-2*	45	20-40	8x8BJ	5.3	70
FK-3	41	20-40	8x8BJ	4.4	145
FK-4	56	20-40	8x8	4.4	140
FK-5	56	20-40	8x8	5.3	70

*-Analytical evaluations performed on these tests

In the CABRI REP Na program, the only tests on UO_2 fuel that experienced cladding failure were rods that contained cladding with spalled outer surface oxide layers. Accompanying the spalled oxide layers in these tests were localized hydride concentrations up to 50% the cladding wall thickness, identified by neutron radiography and post-test metallography. Recent tests REP Na-8 and REP Na-10 failed at fuel enthalpy levels around 60 cal/gm and each rod had an additional energy deposition following cladding failure. Although these test rods contained extensive cladding spallation and hydride localization, only trace amounts of volatile fission products were released into the coolant, no fuel dispersal was experienced, and massive loss of cladding integrity was not observed even though an additional ~30 cal/gm was deposited after cladding failure. CABRI REP Na-8 and Na-10 further support the concept that loss of cladding ductility by localized hydride concentrations is the major cause of cladding failure during rapid energy deposition events at elevated temperatures. Each of these tests exhibited brittle cladding cracks associated with localized hydride concentrations.

Conversely, evaluation of the PCMI-related failures in NSRR shows a strong correlation of the hydride rim thickness and the potential for cladding failure. Post-test examinations on unfailed tests HBO-6 and HBO-7 found part-wall microcracks in the outer surface oxide and hydride rim layer that were blunted in the ductile Zircaloy substrate[3]. Similar microcracks were also observed in the failed tests HBO-1 and HBO-5 in the vicinity of the through-wall cracks. The low initial cladding temperature and the narrow pulse width in the NSRR tests magnify the influence of the hydride rim at the cladding outer surface on the effective cladding ductility. The test conditions used thus far in the NSRR tests produce high cladding stress prior to significant cladding heating from the pellet. At low temperature, the effective ductility of Zircaloy cladding containing non-uniform hydride layers is low and may be insufficient to accommodate fuel pellet expansion.

In the majority of the NSRR tests that resulted in cladding failure, a small amount of fuel dispersal was observed following cladding failure[10]. The presence of fuel dispersal in the NSRR tests is related to the narrow pulse widths used in the RIA simulations. Since almost no heat conduction occurs during the energy deposition in NSRR, the pellet is under a large compressive stress state, particularly in the rim region. Upon cladding failure, the removal of the confinement stress of the cladding causes immediate stress release within the fuel. The release of the high compressive stress in the fuel produces local cracking and expulsion of small particles through the crack opening.

In tests conducted in NSRR and CABRI above 55 GWd/tU, the effect of burnup has been shown to increase the PCMI forces on the cladding by the gradual reduction of the fuel-cladding gap thickness during steady state irradiation. The CABRI REP Na-4 and Na-5 tests at 60+ GWd/tU and the recent NSRR tests FK-4 and FK-5 at 56 GWd/tU did not fail after experiencing maximum fuel enthalpy increases between 70 and 140 cal/gm. The results from these tests demonstrate the absence of any uncontrolled PCMI loading mechanisms associated with burnup related processes such as the pellet rim formation. Even at a uniform oxide level thickness of 80 microns, post-test examinations find that the cladding of CABRI REP Na-4 was able to withstand plastic deformation approaching 0.4% during the rapid energy deposition.

The results from the CABRI tests on MOX fuel irradiated between 28 and 47 GWd/tU show higher fission gas release and an increased PCMI loading from fission gas-induced pellet swelling. CABRI REP Na-7, which had a burnup of 55 GWd/tU, failed during the power pulse although the cladding contained a 50-micron uniform oxide layer thickness. The Na-7 results suggest that a significant contribution of fission gas expansion or pressure loading was applied to the cladding during the power pulse to cause cladding failure. On-going post-test examinations should provide insights to the behavior of high burnup MOX fuel during an RIA event.

The statistical failure/no failure method used to evaluate the data produces mixed results because of factors introduced during irradiation such as the decrease in fuel rod reactivity and the decrease in cladding ductility with fluence and corrosion for older cladding designs. In addition, variations in test conditions make it difficult to directly compare the results from different test programs. To more fully understand the contribution of burnup, cladding embrittlement and test conditions requires analysis of the test rods using sophisticated analytical capabilities to distinguish between the various effects. Using this method, the circumstances leading to fuel rod failure can be defined as a function of the parameters that influence the response of the fuel and cladding.

ANALYSIS OF RIA-SIMULATION EXPERIMENTS USING FALCON

To assist in the interpretation of the results, detailed fuel behavior analyses were performed for key RIA-simulation experiments using the EPRI fuel behavior code FALCON. The RIA-simulation tests analyzed with FALCON are marked by an asterisk in Tables 1 through 3. FALCON is a two-dimensional transient fuel behavior program developed to analyze the response of LWR fuel rods during RIA, LOCA and other transient conditions[11]. FALCON, which is an improved version of FREY, utilizes a coupled thermal and mechanical finite element methodology to represent the transient behavior of the fuel, cladding and gap[12]. A complete fuel pellet mechanical constitutive model is used that includes pellet cracking, creep, plasticity, and thermal expansion. The effects of burnup and fast neutron fluence are included in the thermal and mechanical properties of the fuel and cladding, as well as, on the radial power distribution. Recently, modifications have been incorporated into FALCON to calculate the sodium coolant temperature heatup during the experiment. Earlier analyses used estimated heat transfer coefficients and coolant temperatures to model the fuel-sodium heat transfer.

Sodium temperature thermocouple measurements, in-pile cladding axial elongation and residual cladding hoop strains have been used to validate the FALCON capabilities for RIA analyses. A comparison of the thermocouple response at 48 cm (near the top of the fuel stack) and the calculated sodium temperature at different locations in the FALCON model is shown in Figure 1 for the CABRI REP Na-4 test. The peak sodium temperature measured by the two thermocouples located 120° apart varied from 373° C to 396° C which agrees well with the calculated sodium temperatures at 46 cm (383° C) and 52 cm (392° C). The cladding axial elongation response measured using an in-pile LVDT device is shown in Figure 2 for

CABRI REP Na-4. Also shown for comparison is the calculated cladding elongation. FALCON calculates a slightly lower peak cladding elongation (3.7 mm versus 4.2 mm), however, the overall trend in the cladding elongation response after the power pulse is represented well.

A comparison of the calculated and measured axial profile of the cladding radial displacements for REP Na-5 is shown in Figure 3. Two different azimuthal traces are shown to indicate the variation in the measured data. Excellent agreement is seen though out the entire axial length of the test specimen. FALCON has a tendency to calculate slightly higher cladding radial displacements at the end of the test specimen. This could be due to uncertainty in the axial power shape used in the analysis.

Figures 4 and 5 show a comparison between the measured and predicted residual cladding hoop strains (Figure 4) for the CABRI REP Na tests and the measured and predicted cladding and fuel axial elongation for the CABRI and NSRR tests analyzed with FALCON (Figure 5). The calculated and measured cladding hoop strain results shown in Figure 4 represent the mid-pellet strain at the maximum power location. The error bars shown in Figure 4 indicate the variation in the measured data based on the measurement uncertainty. Good agreement is observed between the measured and calculated radial and axial cladding deformations.

These results demonstrate the capability of FALCON to properly model the complex thermal and mechanical behavior of high burnup fuel during rapid energy depositions corresponding to a RIA event. As summarized in Tables 1 through 3, the validation of FALCON for RIA analyses includes a wide variety of pulse characteristics, coolant conditions, and fuel rod types.

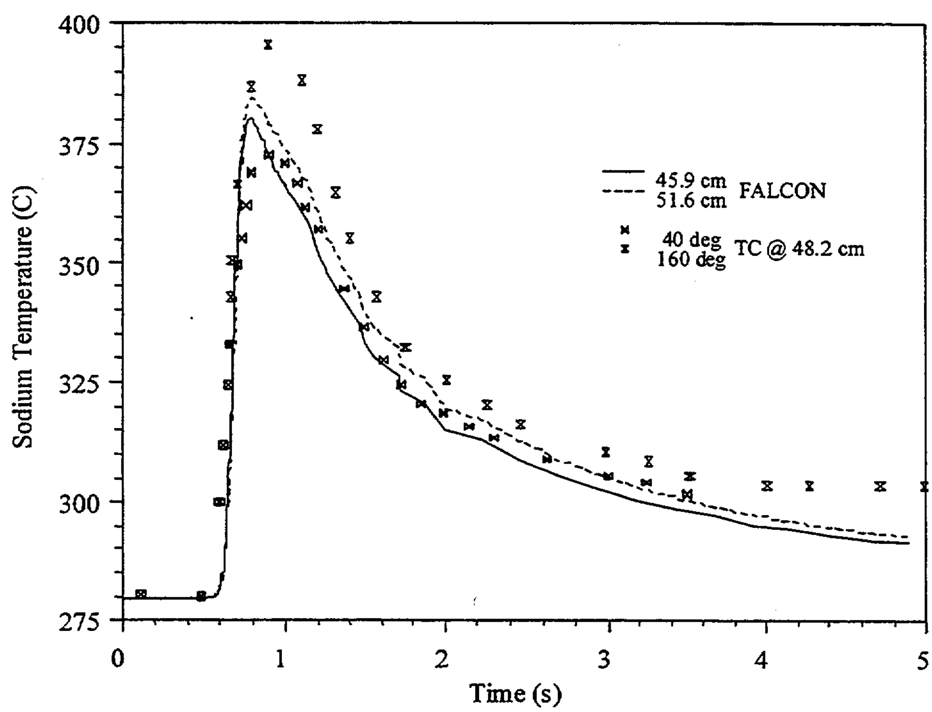


Figure 1. Sodium Coolant Temperature during CABRI REP Na-4

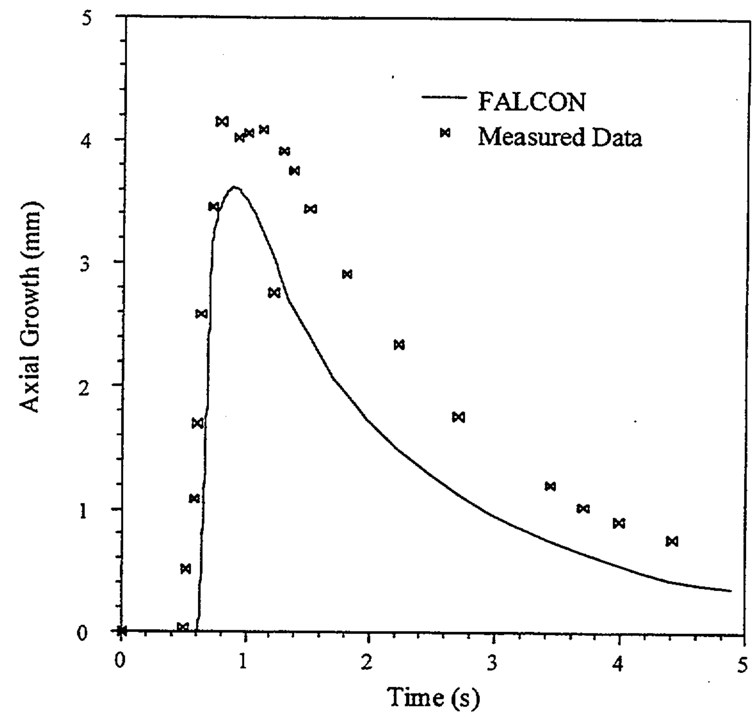


Figure 2. FALCON Results and Measured Data for the Time History of the Cladding Axial Elongation in CABRI REP Na-4

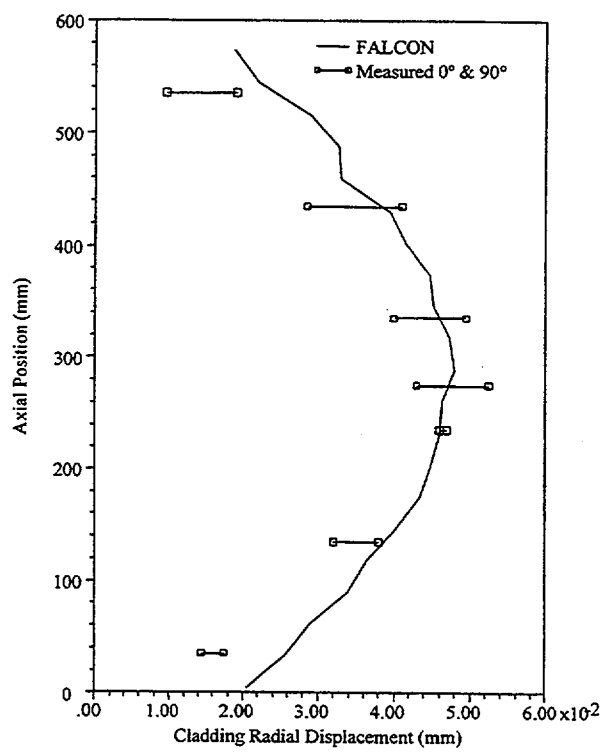


Figure 3. Comparison of FALCON and Measured Residual Cladding

Radial Displacements for CABRI REP Na-5

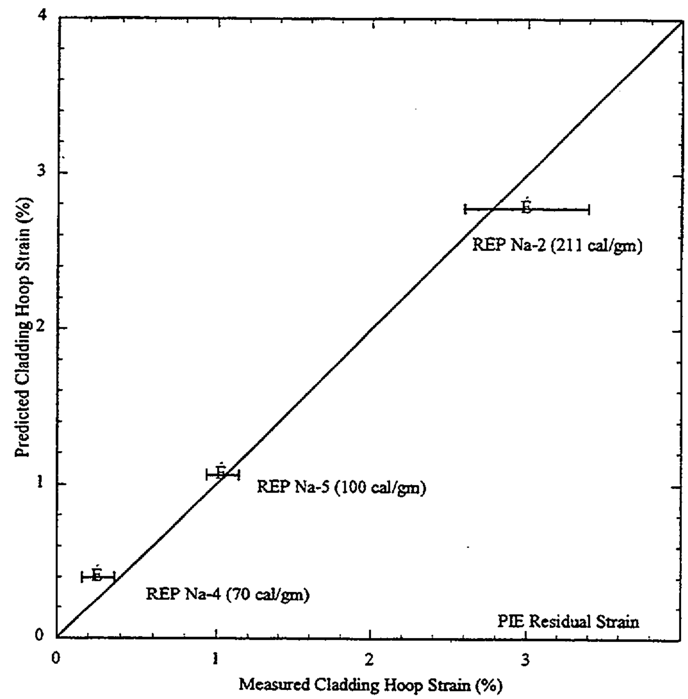
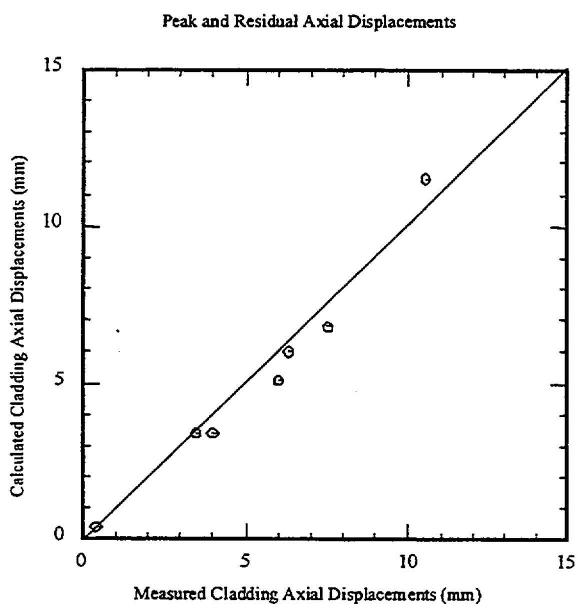
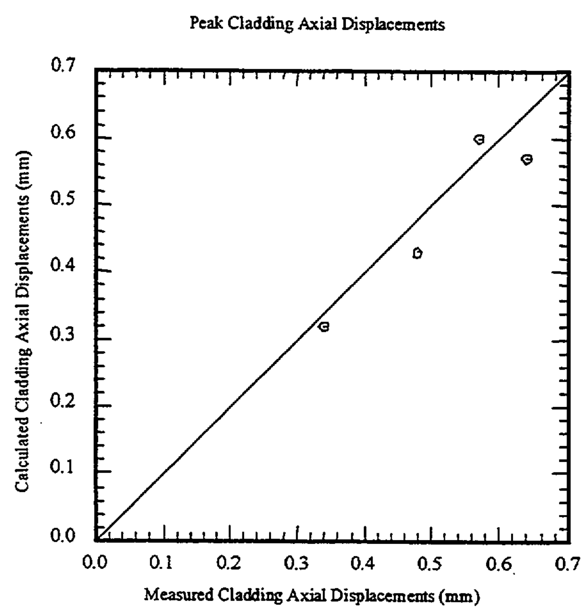


Figure 4. Predicted versus Measured Residual Cladding Hoop



CABRI REP Na Test Results



NSRR HBO and JM Tests
SPERT-CDC Tests

Strain for Selected CABRI REP Na Tests

Figure 5. Predicted versus Measured Residual Cladding Hoop Strain for Selected CABRI and NSRR tests analyzed with FALCON

Cladding Integrity Model

Part of the analytical evaluation included the calculation of the cladding failure potential caused by PCMI during the RIA-simulation experiments. The assessment of cladding failure requires a method to evaluate the material limit under the loading and temperature conditions of the accident event. The methods available include strain-based criteria such as total or uniform elongation, fracture mechanics, and strain-energy based concepts[13]. Each method requires mechanical property data to develop the material limit as a function of temperature, irradiation and material condition. A cladding integrity model based on the Strain Energy Density (SED) approach was developed for use during the power pulse of an RIA event when PCMI forces dominate[14]. It is envisioned that the SED methodology will prove to be more versatile than a strain or fracture mechanics approach and can address the influence of strain rate, material plasticity, and multi-stress states. The SED approach consists of two components: the Critical SED (CSED) and the SED. The CSED is a measure of the cladding material failure limit and is developed based on mechanical property data. The SED is a measure of the loading intensity on the cladding and is calculated using FALCON. The Calvert Cliffs, ANO-2, and PROMETRA mechanical property tests on irradiated cladding material were used to develop the CSED curve[15,16,17]. The CSED values from the tests are shown in Figure 6 for temperatures above 280°C and in Figure 7 for temperatures below 150°C. Also included are the least-squares best-fit relations. To facilitate the development of the CSED curve, the data above 280°C were divided into two datasets that reflect the material condition: one for samples without oxide spallation and one for samples with oxide spallation. Separate least-square curve fits were performed for each dataset. The results shown in Figure 6 indicate a significant variation between the non-spalled oxide samples and those containing oxide spallation.

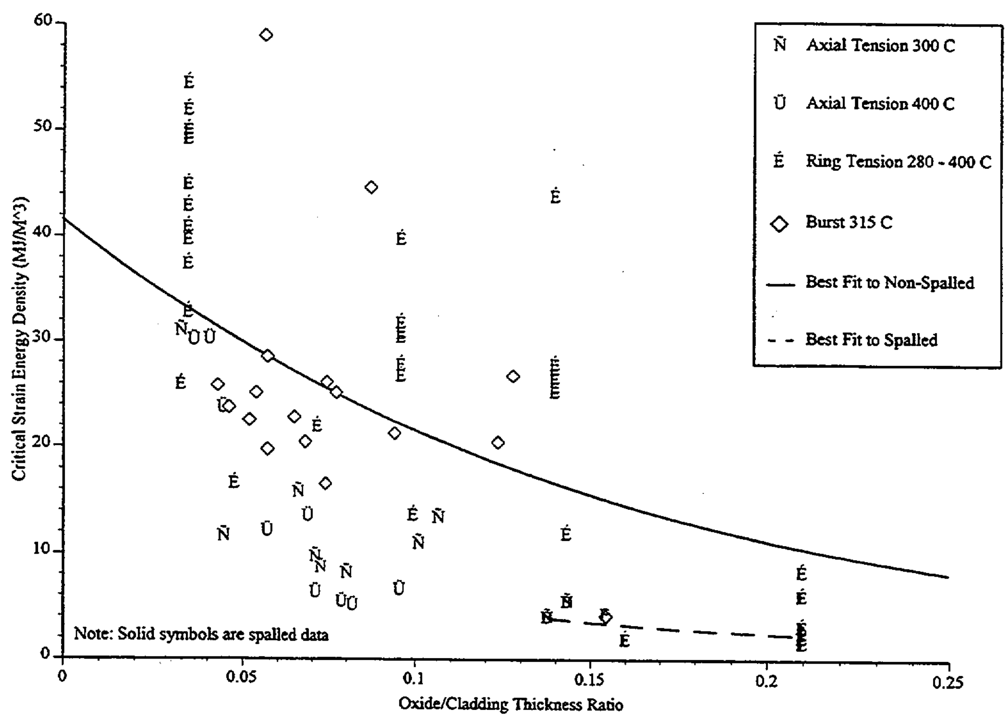


Figure 6. Critical Strain Energy Density Values from Mechanical Property Tests at Temperatures Above 280°C. Best-fit Curves of the Non-Spalled and Spalled Data are Shown for Comparison.

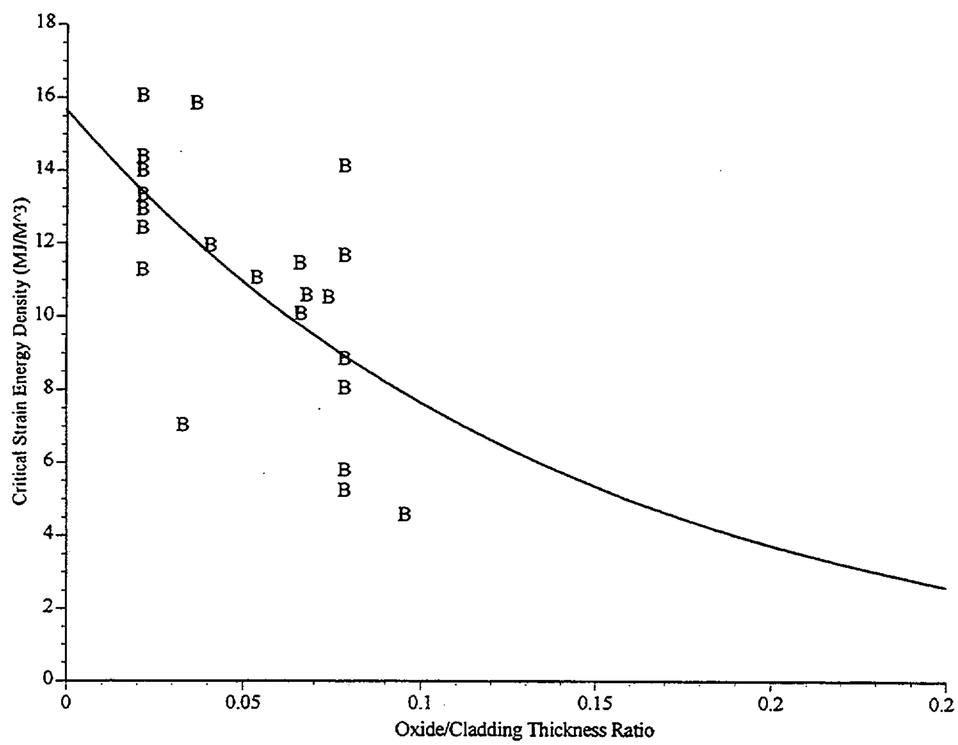


Figure 7. Critical Strain Energy Density Values from Mechanical Property Tests at Temperatures Below 150°C. Best-fit Curve of the Data is Shown for Comparison.

The SED throughout the cladding is calculated during the PCMI phase for each RIA-simulation test analyzed with FALCON. The SED model in FALCON performs a summation of the product of the stress and strain increment for the three coordinate stresses and strains (radial, axial, and hoop) at every location in the cladding. This quantity is then integrated over the time of the transient to provide the SED (U(t)) as a function of time. The maximum SED at the time corresponding to the end of the power pulse or the time of cladding failure determined by in-pile instrumentation is shown in Figure 8 for the sodium tests (T > 280°C) and Figure 9 for the stagnant water tests (T < 150°C). Also shown for comparison are the appropriate best-fit CSED curves from Figures 6 and 7. The tests in sodium coolant on rods with non-spalled oxide layers lie below the CSED curve for non-spalled cladding, indicating a low potential for cladding failure. However, the two tests on rods with spalled oxide layers (REP Na-8 and Na-10) reside either on or above the CSED curve for spalled cladding, indicating a high potential for cladding failure. Similarly in Figure 9, the SED results for the NSRR tests that did not fail reside below the CSED curve for low temperature conditions. Tests that experienced failure or contained micro-cracks in the outer region of the cladding reside near or above the CSED curve. Based on these results, the coupled FALCON/SED model displays a reasonable level of success in separating failed and non-failed tests using a CSED limit developed from cladding mechanical property tests on irradiated cladding.

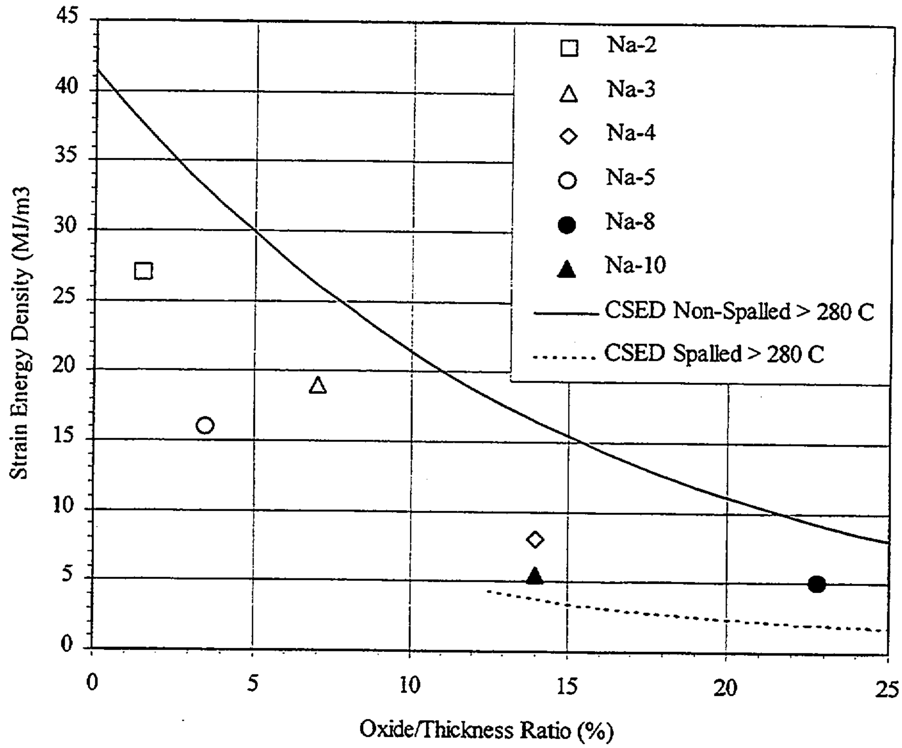


Figure 8. Strain Energy Density Results Calculated by FALCON for the CABRI REP Na Tests on UO_2 Test Rods. CSED Model is Shown for Comparison.

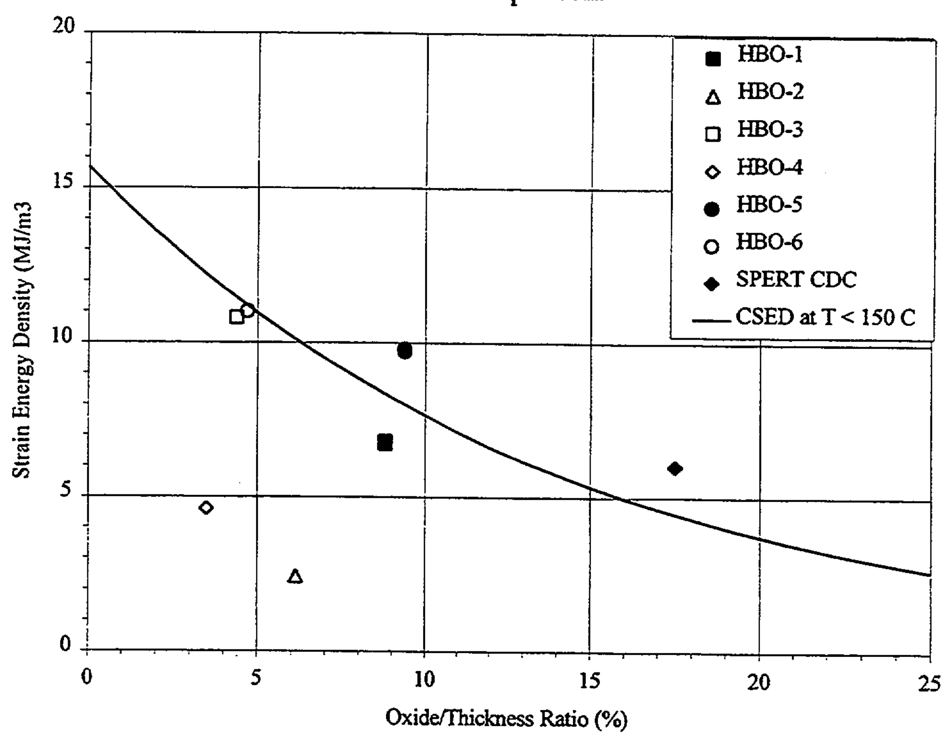


Figure 9. Strain Energy Density Results Calculated by FALCON for Selected NSRR Test Rods. CSED Model is Shown for Comparison.

Analysis of the CABRI and NSRR tests find that an analytical translation is required to correct for the variation in the initial coolant conditions before a comparison of the fuel enthalpy levels is performed. The FALCON methodology provides for this translation through the calculation of the cladding SED response, and combined with CSED at the appropriate temperature condition, establishes a basis for a comparison. Furthermore, the analytical evaluation suggests that the CABRI tests performed in high temperature sodium represent well the PCMI phase of a postulated in-reactor RIA event and require only limited analysis to translate the results to PWR conditions. On the other hand, the NSRR tests conducted in room temperature stagnant water require analytical translation to PWR conditions before the results can be used to evaluate the fuel rod failure licensing bases.

CONSIDERATIONS FOR RIA LICENSING CRITERIA

The current fuel rod failure and core coolability limits are defined in NUREG-0800 and NRC Regulatory Guide 1.77 for a Control Rod Ejection Accident (CREA) in a PWR and Control Rod Drop Accident (CRDA) in a BWR. Fuel rod failure is assumed to occur at the initiation of DNB or at a radially averaged fuel enthalpy of 170 cal/gm at any axial location. The core coolability (violent expulsion of fuel) limit is specified to be a maximum radially averaged fuel enthalpy of 280 cal/gm. These limits were developed based on RIA-simulation tests performed on test rods with zero or low burnup, thus minimizing complications introduced during burnup accumulation.

The core coolability limit represents the absolute safety limit and must not be exceeded during fuel reload licensing calculations for a hypothetical accident. The fuel failure limit, on the other hand, can be exceeded for a specific number of fuel rods and is used to define the radiological source term for comparison to off-site dose limits

Figure 10 contains a schematic of the RIA fuel failure and core coolability limits defined in terms of fuel enthalpy increase as a function of some parameter that represents the fuel or cladding condition. Examples of this parameter include burnup and oxide thickness. The coolability and fuel rod failure limits remain separated throughout the entire range of applicability.

Experimental data on high burnup test rods REP Na-8 and REP Na-10 support the separation of the fuel failure and fuel dispersal enthalpy levels. Relevant data indicate that the fuel enthalpy difference between fuel failure and fuel dispersal can be greater than 30 cal/g at burnup levels up to 64 GWD/T with oxide thicknesses greater than 100 microns with spallation and pulse widths as narrow as 30 milliseconds. As fuel dispersal does not necessarily lead directly to core coolability concerns such as pressure pulses or flow blockage in LWRs, the core coolability limit may reside well above this level of fuel enthalpy.

The schematic in Figure 10 contains two different plateaus. The high plateau at low values of the dependent parameter considers fuel failure and coolability controlled by high temperature mechanisms such as fuel and clad melting. The low plateau at higher values of the dependent parameter considers PCMI controlled fuel failure and coolability. For licensees to demonstrate compliance to limits like that shown in Figure 10 may require the use of multi-dimensional neutron kinetics methods. Furthermore, these methods may require the use of assumptions commensurate with the risk to provide a realistic estimate of the fuel enthalpy levels during a RIA event.

Data Needs for Burnup Extension

The current understanding of fuel behavior during energy deposition for a RIA event extends to 65 GWd/tU peak nodal burnup. However, there is interest to increase peak nodal burnup levels to 70 - 75 GWd/tU as part of an industry effort to extend fuel assembly burnup goals through longer fuel cycles and higher energy utilization. To achieve this goal, additional data is required to further the understanding of high burnup fuel RIA behavior.

In the CABRI tests, the MOX tests exhibit higher fission gas release and large cladding deformations as compared to the UO_2 tests. These results suggest a strong contribution of gaseous swelling to the PCMI loading mechanisms in the MOX tests. Furthermore, the behavior of a high burnup MOX test (55 GWd/tU) displays evidence of cladding failure assisted by fission gas pressure. Since the pellet rim of high burnup UO_2 fuel displays similar characteristics of MOX fuel, the MOX test results may be an indication of fuel behavior for UO_2 fuel irradiated to burnup levels greater than 70 GWd/T. This indicates a possible need for additional tests beyond 70 GWd/tU to verify the stable performance of UO_2 fuel.

Separate effects tests are required to gain a better understanding of the transient fission gas release kinetics and the mechanical behavior of advance cladding alloys. Additional data on the fission gas release kinetics can provide insights into the contribution that fission gases have on the PCMI loading during the power pulse. This information can be incorporated into the analytical methods used to interpret the experiments and to perform licensing calculations. Similarly, mechanical property tests on advanced cladding alloys can be used to develop improvements to the CSED model used to evaluate cladding integrity. An improved CSED model can be used to translate the RIA data and criteria developed for Zircaloy to various advanced alloys. Such tests should consider the effects of stress state, temperature and loading rate.

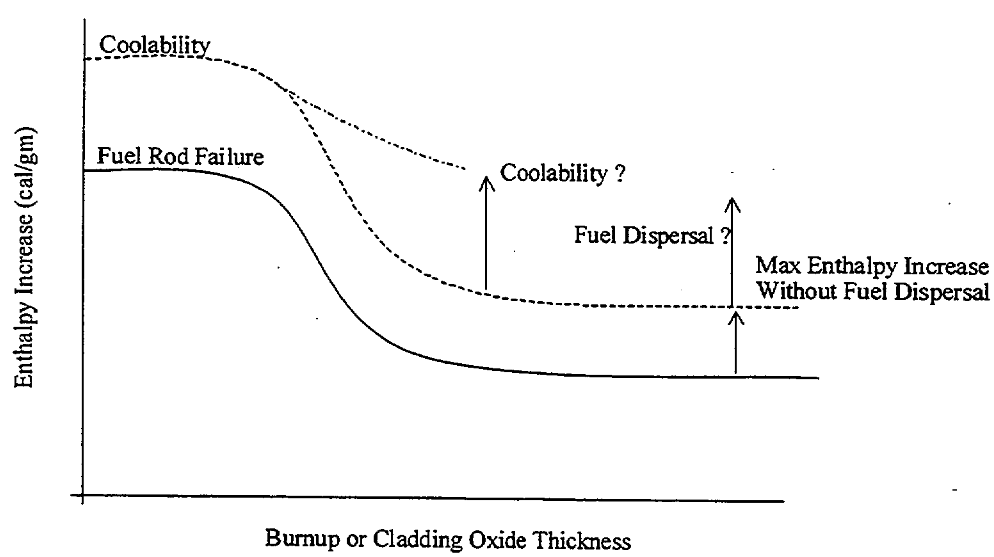


Figure 10 . Schematic Diagram of the RIA Fuel Failure and Core Coolability Limits.

CONCLUSIONS

The industry has adopted a coupled probabilistic and deterministic approach to resolve the RIA issue. The PSA evaluations will be used to guide the R&D program and define the safety criteria. The deterministic evaluations performed have led to the following conclusions:

Data Consistency and Analytical Tool. The RIA-simulation tests performed in CABRI and NSRR form a consistent and conservative database. The analytical tool, FALCON, is able to explain key experimental results well; it is also a valuable tool that can be used to cross-compare results from different programs or assess the fuel behavior under different accident scenarios or different materials.

Fuel Failure. For high burnup UO_2 fuel, the cladding failure mechanism is PCMI (pellet cladding mechanical interaction) assisted by loss of cladding ductility from hydride embrittlement of the cladding. For the representative test conditions in CABRI, no fuel failures occurred up to 64 GWD/T as long as the rods did not exhibit oxide spallation. However, failures have occurred on test rods without oxide spallation in NSRR under non-representative test conditions (25 degree C, in stagnant water).

Fuel Dispersal. Experimental data support the separation of the fuel failure and fuel dispersal enthalpy levels for the entire burnup range. Relevant data indicate that the fuel enthalpy difference between these two levels can be greater than 30 cal/gm for high burnup fuel (up to 64 GWD/T). As fuel dispersal does not necessarily lead directly to pressure pulses or flow blockage in LWRs, the core coolability limit may reside well above this level of fuel enthalpy.

The comprehensive deterministic evaluations, including RIA-simulation tests, separate effects data, neutron kinetic analyses and a good analytic tool like FALCON, combined with a PSA will provide a strong basis to resolve the RIA issue for current fuel designs and burnup extensions.

REFERENCES

1. Chung, H.M., L.A. Neimark, and T.F. Kassner, "Test Plan for High-Burnup Fuel Behavior Under Loss-of-Coolant Conditions," *Proceedings of the Twenty-Fourth Water Reactor Safety Meeting*, Vol. 1, Nuclear Regulatory Commission, Bethesda, Maryland, October 21-23, 1996, pp. 173-184.
2. Papin, J., et al., "French Studies on High-Burnup Fuel Transient Behavior Under RIA Conditions," *Nuclear Safety*, Vol. 37, No. 4, Nuclear Regulatory Commission, Bethesda, Maryland, October 21-23, 1996, pp. 289-327.
3. Fuketa, T., et al., "NSRR/RIA Experiments with High-Burnup PWR Fuels," *Nuclear Safety*, Vol. 37, No. 4, Nuclear Regulatory Commission, Bethesda, Maryland, October 21-23, 1996, pp. 328-342.
4. Schmitz, F., et. al., "Investigation of the Behavior of High Burnup-Up PWR Fuel Under RIA Conditions in the CABRI Test Reactor," *Proceedings of the Twenty-Second Water Reactor Safety Meeting*, Vol. 1, Nuclear Regulatory Commission, Bethesda, Maryland, October 24-26, 1994, pp. 107-108.
5. Fujishiro, T. and K. Ishijima, "NSRR Experiments to Study the Effects of Burnup on the Fuel Behavior Under Reactivity-Initiated Accident Conditions, " *Proceedings of the Twenty-Ssecond Water Reactor Safety Meeting*, Vol. 1, Nuclear Regulatory Commission, Bethesda, Maryland, October 24-26, 1994, pp. 109-110.
6. O.Ozer, Y.R. Rashid, R.O. Montgomery, R.Yang, "Evaluation of RIA Experiments and Their Impact on High Burnup Fuel Performance," *Proceedings of the CSNI Specialists Meeting on Transient Behavior of High Burnup Fuel*, Cadarache France, September 12-14, 1995, Report NEA/CSNI.R(95) 22, 1996.

7. Montgomery, R.O. and Y.R. Rashid, "Evaluation of Irradiated Fuel During RIA-Simulation Tests", EPRI TR-106387, May 1996.
8. Montgomery, R.O., et al., "Review and Analysis of RIA-Simulation Experiments on Intermediate and High Burnup Test Rods," *Proceedings of the ANS International Topical Meeting on Light Water Reactor Fuel Performance*, Portland, Oregon, March 2-6, 1997 pp. 711-720.
9. Schmitz, F. and J. Papin, "REP-Na 10, another RIA Test with a Spalled High Burnup Rod and a with Pulse Width of 30 ms," *Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting*, Vol. 3, Nuclear Regulatory Commission, Bethesda, Maryland, October 26-28, 1998, pp. 243-254.
10. Fuketa, T, et al., "NSRR Pulse Irradiation Experiments and Tube Burst Tests," *Proceedings of the Twenty-Sixth Water Reactor Safety Information Meeting*, Vol. 3, Nuclear Regulatory Commission, Bethesda, Maryland, October 26-28, 1998, pp. 223-242.
8. "FALCON – Fuel Analysis and Licensing Code, Volume 2: User's Manual", ANA-97-0230, Anatech Corporation, San Diego, California, December 1997.
12. Rashid, Y.R., R.O. Montgomery, and A.J. Zangari, "FREY-01: Fuel Rod Evaluation System, Volume 1: Theoretical and Numerical Bases", EPRI NP-3277-V1R3, August 1994.
13. Wnuk, M.P. and A. Mazur, "Experimental Verification of the Strain Energy Density Criterion for the Stable Phase of Tensile Fracture," *Proceedings of and International Symposium on Absorbed Specific Energy and/or Strain Energy Density Criterion*, Budapest, Hungary, September 17-19, 1980 pp. 297-307.
14. Sih, G.C., *Mechanics of Fracture Initiation and Propagation – Surface and Volume Energy Density Applied as Failure Criterion*, Kluwer Academic Publishers, Dordecht, The Netherlands, 1991.
15. Smith, G. P. et al., "Hot Cell Examination of Extended Burnup Fuel from Calvert Cliffs-1", Vol. 2, EPRI TR-103302-V2, Electric Power Research Institute, Palo Alto California, July 1994.
16. Smith, G. P. et al., "The Evaluation and Demonstration of Methods for Improved Nuclear Fuel Utilization", DOE/ET/34013-15, U.S. Department of Energy, Washington, D.C., August 1994.
17. Lemoine, F., and M. Balourdet, "RIA Related Analytical Studies and Separate Effects Tests," *Proceedings of the ANS International Topical Meeting on Light Water Reactor Fuel Performance*, Portland, Oregon, March 2-6, 1997 pp. 693-703.

MECHANICAL PROPERTIES OF UNIRRADIATED AND IRRADIATED ZR-1%NB CLADDING UNDER ACCIDENT CONDITIONS*

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Abstract

The experimental program aimed to obtain the modern data base with mechanical properties of Zr-1%Nb cladding alloy is overviewed, and test results for unirradiated and highly irradiated claddings of commercial fuel rods are presented. The structure of test program included two types of mechanical tests which were conducted within the temperature range 20 – 1200°C in order to cover both low temperature loading conditions, and post-DNB conditions related to reactivity initiated accident (RIA). Uniaxial tensile tests in a transverse direction allowed to obtain the basic stress-strain relations for modeling the cladding plastic deformation. Biaxial tube burst tests were performed to study cladding failure parameters and to verify the uniaxial results. On the basis of uniaxial and biaxial data, the cladding failure criteria are proposed. The results of recent low temperature (20 – 450°C) burst tests are reported. The tests were carried out for two values of the biaxiality ratio, 2 and 1, that represent characteristic stress states of the cladding during pellet-cladding mechanical interaction.

Introduction

The burnup level of 50–60 MWd/kg U that can be reached in the commercial reactors made it necessary to pay attention to the safety criteria for the fuel rods of the light water reactors under accident conditions. Now, the wide range of investigations is being performed in this line. Modification and improvement of the database on the fuel rod cladding mechanical properties in accordance with the requirements of the computer codes intended to analyze mechanical behavior of the fuel claddings under accident conditions is an important component of this work. Reassessment of the data bases that are being currently used has indicated a number of drawbacks related with deficiency of material properties of highly irradiated cladding of the commercial reactors.

Therefore, a special program was initiated to get a modern data base to analyze mechanical behavior of Zr-1%Nb claddings under accident conditions. The main objective of the investigation was to obtain short-term mechanical properties and parameters of the cladding failure, which were later to be included into the computer libraries of material properties. The analysis of reliability of the calculated results obtained with the help of the developed correlations of the material properties was another objective of the investigations.

General overview of the test program

The structure of the test program is schematically presented in Fig. 1. Two important areas were considered during the development of the program from the point of view of application of test results to the analysis of fuel rod behavior:

- low temperature cladding deformation and failure due to pellet-cladding mechanical interaction (PCMI) at the early stage of reactivity initiated accident;
- high temperature rupture of the cladding (HTRC) due to internal gas pressure at the post-DNB stage of the accident.

According to these problems, tests were directed to obtain both the basic mechanical properties and failure criteria appropriate for various accident conditions.

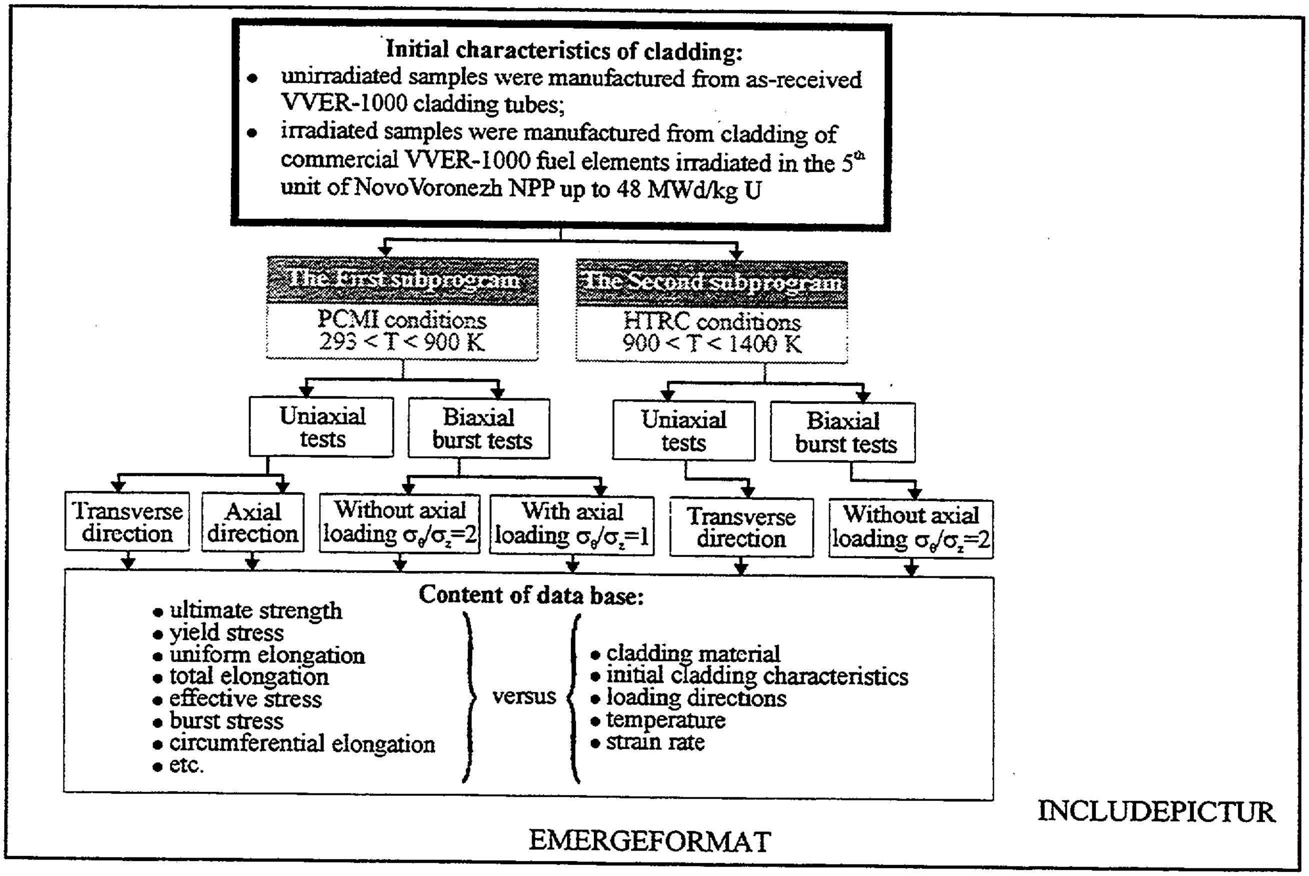

Fig. 1. The structure of the test program on mechanical properties of Zr-1%Nb cladding.

Within the framework of this program, we analyzed critically the archive results of the measured mechanical properties of the unirradiated claddings. As a consequence, a new data base was obtained for the unirradiated tubes, and for the claddings irradiated in the commercial reactor of the VVER-1000 type up to the burnup of ~50 MWd/kg U. The data base includes the results of the uniaxial tensile tests in a transverse direction, and of the biaxial burst tests of the pressurized tube samples. To simulate a stress state of the cladding at the PCMI, the important efforts were made to perform a low temperature burst testing with an additional value of the stress biaxiality ratio equals to 1. Principal explanations for the scope of the test program are presented in Fig. 2.

Uniaxial tensile properties were obtained with the simple ring specimens according to the standard Russian procedure. A number of problems associated with the interpretation of raw load-displacement curves of that type of specimens was revealed. The question of whether the obtained data can be adequately used to calculate the stress-strain conditions of the cladding has not been answered yet despite the significant update of the previously existed procedure.

A special approach has been worked out to solve this problem. The idea of the approach is to compare the results of ring tensile tests and biaxial burst tests that quite closely repeat the geometry and loading conditions of the real cladding. This comparative analysis required the performing of new tests of both types.

Additional ring tests of unirradiated and irradiated cladding specimens were performed. Along with the measurement of the traditional engineering parameters of strength and ductility, it was also necessary to determine the true stress at rupture. A cross-area reduction procedure was specifically developed for these purposes. On the other hand, a series of burst tests was performed in the temperature range of 20 – 450°C. Engineering ultimate strength and true burst stresses were reviewed as the major results of the tests. Unirradiated and irradiated claddings were loaded by the pressure of liquid at the rate of 1 MPa/s under the constant set temperature. A number of nondestructive and destructive post-test procedures was performed for the quantitative definition of stresses and strains in the specimens. For the correct comparison, the strength parameters obtained from those two types of tests were converted into the effective stresses with the help of the existing data base on the anisotropy coefficients of Zr-1%Nb claddings.

The series of burst tests was also important to determine cladding failure parameters under low temperatures typical for the PCMI stage of the reactivity initiated accident. Although the majority of burst tests were aimed at the LOCA problems, there is practically no experimental data base to develop the failure criterion in the low-temperature region. Within the framework of this task along with the typical tests with the loading by internal pressure only, a number of parallel tests was performed with additional axial load simulating axial constraining of the cladding by the expanding fuel in the case of PCMI. The law of the change of the axial force was specified to support equality between tangential and axial stresses in the specimen within the whole duration of the test. Therefore, testing of unirradiated and highly irradiated claddings was performed for two values of the ratio of biaxiality - 2 and 1 that covers the expected range for the reactor PCMI case.

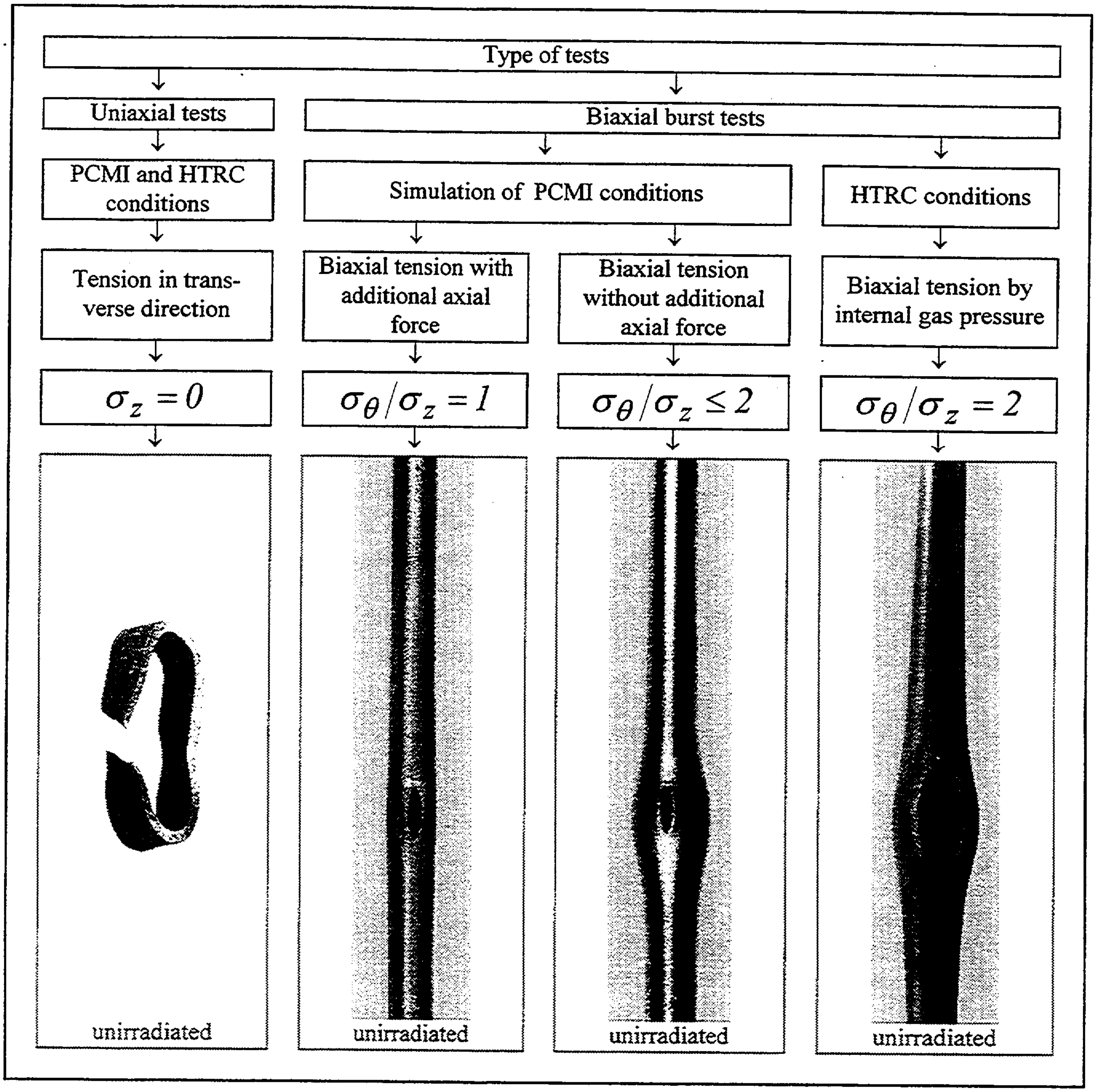

Fig. 2. The scope of the test program.

Results

A comprehensive description of the procedures and results of uniaxial tensile tests at temperature 293 – 1223 K as well as of high temperature biaxial tube burst tests is given in [1]. The major results of these types of tests, which have been performed at the first steps of the program execution, are as follows.

Uniaxial tests

Ultimate tensile strength, yield stress, uniform elongation, and total elongation for unirradiated and irradiated cladding plotted versus temperature are presented in Fig. 3, Fig. 4. Presented data were obtained at the basic strain rate 0.002 1/s. The influence of strain rate over the strength and ductility was studied within the strain rate range 0.002 – 0.5 1/s for the entire temperature interval. The summary of these studies is presented in Fig. 5 as the temperature dependence of the stain rate sensitivity exponent *m*.

The following conclusions could be made on the basis of obtained uniaxial data:

- The strength of cladding material decreases as the temperature increases while the ductility parameters are rather insensitive to the temperature increase up to 900 K.
- Base irradiation results in the higher strength and lower ductility, however, at temperatures above 900 K there are no important differences between mechanical properties of unirradiated and irradiated material.
- The strain rate sensitivity is relatively low at low temperatures and sharply increases at temperatures above 800 K.
- Ductile-to-brittle material transition was not observed for unirradiated and irradiated cladding in the tested range of temperatures and strain rates.

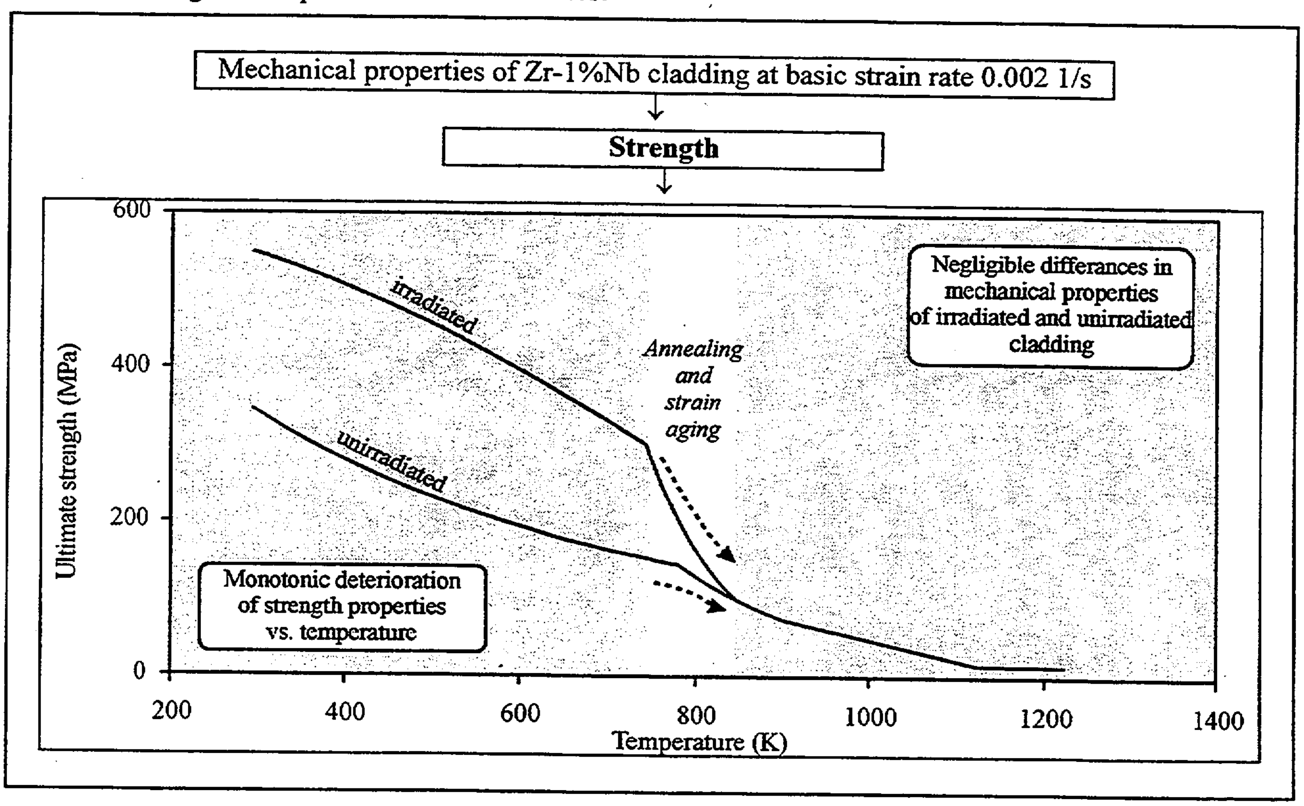

Fig. 3. Results of uniaxial tests.

High temperature burst tests

Biaxial burst tests of pressurized tube samples were conducted within the temperature range 973 – 1473 K. Unirradiated and irradiated cladding samples were loaded by internal gas pressure with pressurization rates 0.01 – 1.0 MPa/s under isothermal conditions. The environment was inert in all tests. The coincidence of both strength and ductility parameters was observed for unirradiated and irradiated materials. The results on the maximum circumferential elongation at burst are presented in Fig. 6. The post-test measurements of cladding shape and destructive examinations allowed to estimate a true hoop stress at burst, which was accepted as a failure criterion for the high temperature rupture of ballooning type. This criterion is plotted in Fig. 7 along with the analogous estimates of the burst stress from in-pile test data [1]. Both the uniaxial stress-strain data and high temperature failure criterion were incorporated into the U.S. NRC FRAP-T6 code [2] and into the SCANAIR (IPSN, France) code [3]. The VVER-adapted versions of the codes were used to analyze the fuel rod behavior under pulse tests performed in the IGR reactor [1].

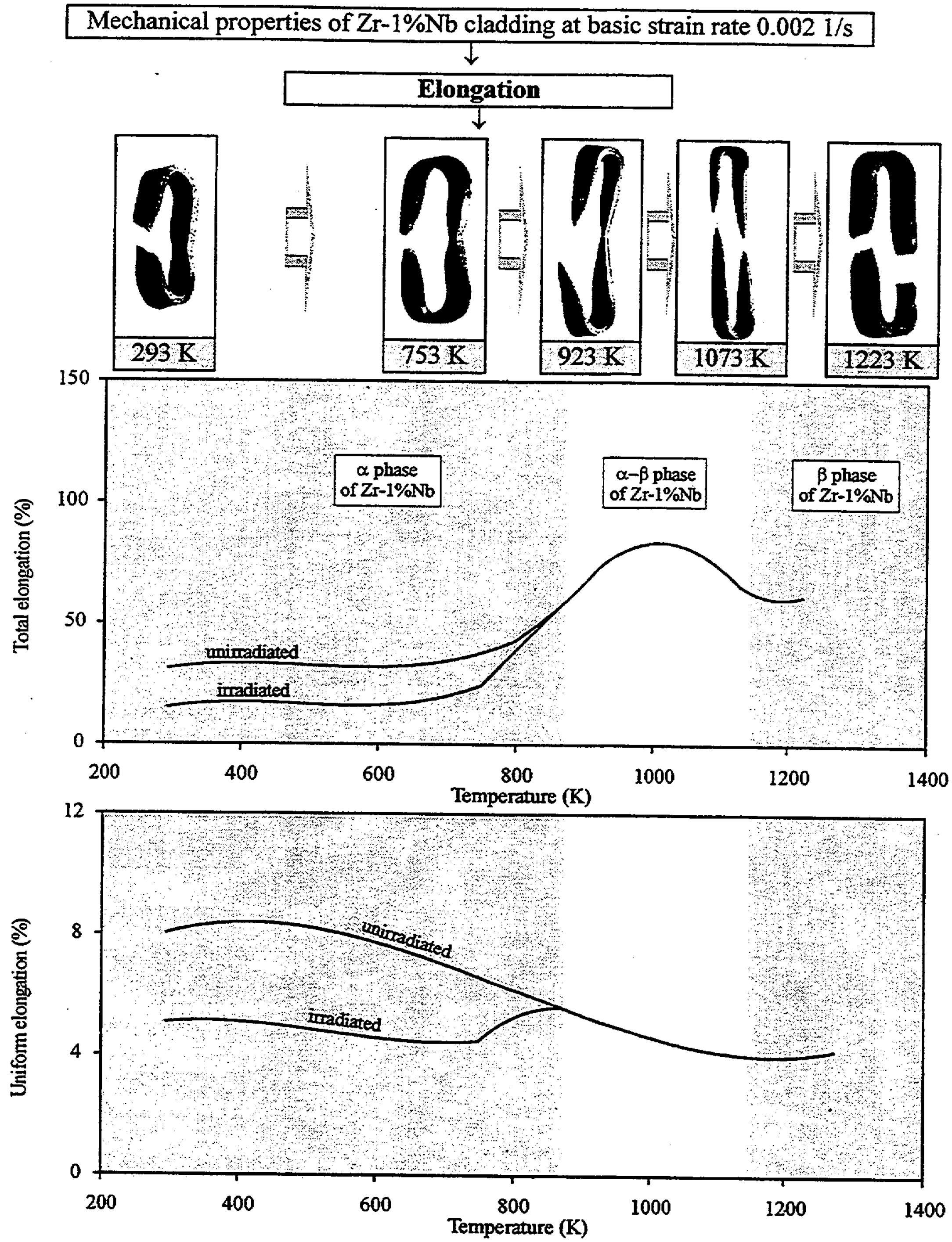

Fig. 4. Total elongation and uniform elongation vs. temperature for unirradiated and irradiated Zr-1%Nb cladding at the strain rate $2 \cdot 10^{-3}$ s^{-1}.

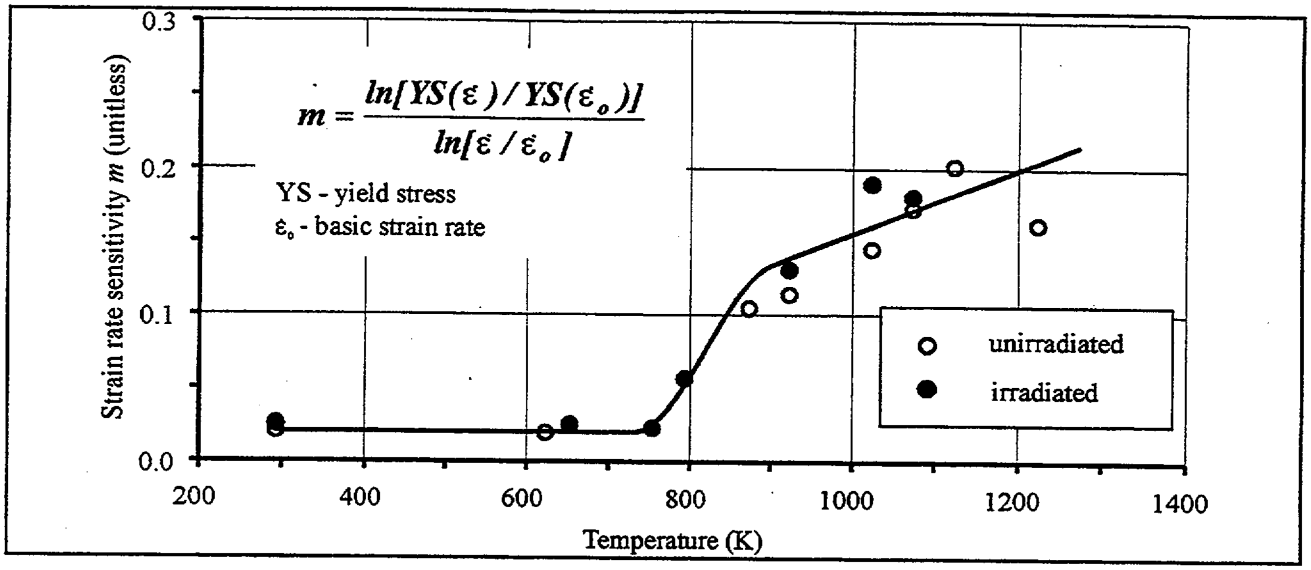

Fig. 5. Strain rate sensitivity exponent vs. temperature.

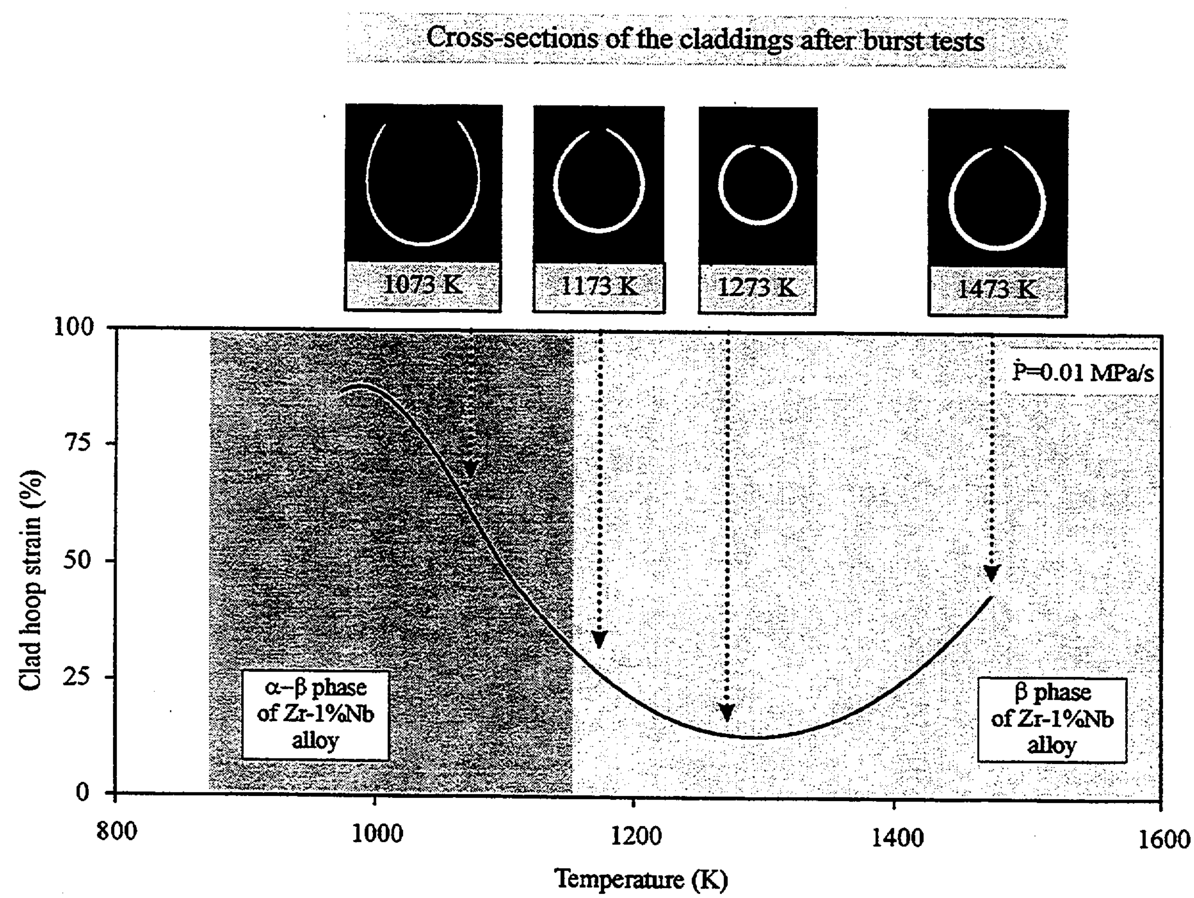

Fig. 6. Circumferential elongation at burst vs. temperature.

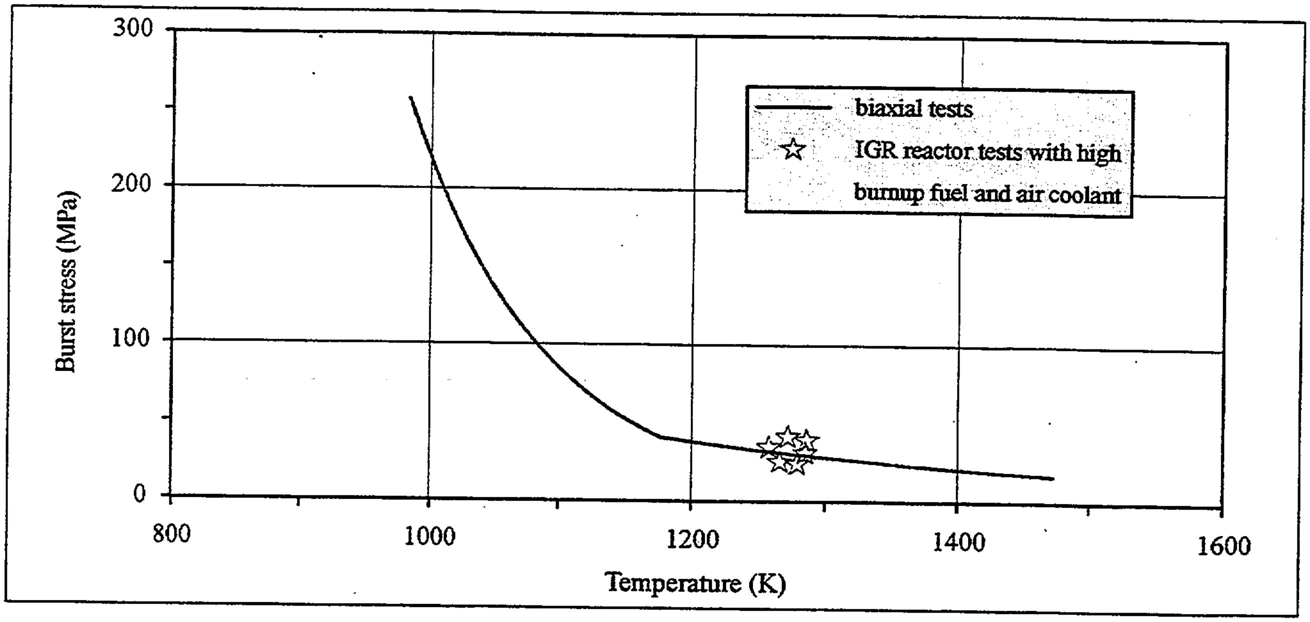

Fig. 7. True hoop burst stress vs. temperature for unirradiated and irradiated cladding.

Low temperature burst tests. Comparison of uniaxial and biaxial data

The results on engineering ultimate strength derived from unirradiated and irradiated burst test data are shown in Fig. 8. The correlations of ultimate strength previously obtained in ring tensile tests are also presented. A good agreement between the results of uniaxial and biaxial tests suffers to conclude that ring tests provide the reliable data intended to model the deformation behavior of the cladding. It should be noted that the comparison of true effective stresses derived from the tests of both types is now under completing. Important results on elongation were drawn during the tests. Fig. 9 presents the maximum circumferential elongation for unirradiated and irradiated tube samples tested with various biaxiality ratios. As it was expected, axial constraining significantly reduces the hoop strain at burst in case of unirradiated cladding. This effect was not observed for irradiated cladding: the maximum hoop strain remains practically the same at axial constraining with the stress biaxiality about 1. Since the anisotropy of mechanical properties of highly irradiated cladding is still rather clouded problem, the obtained insensitivity of elongation to biaxiality ratio can be considered as the important evidence of the fact that the irradiated cladding material is close to the isotropic one.

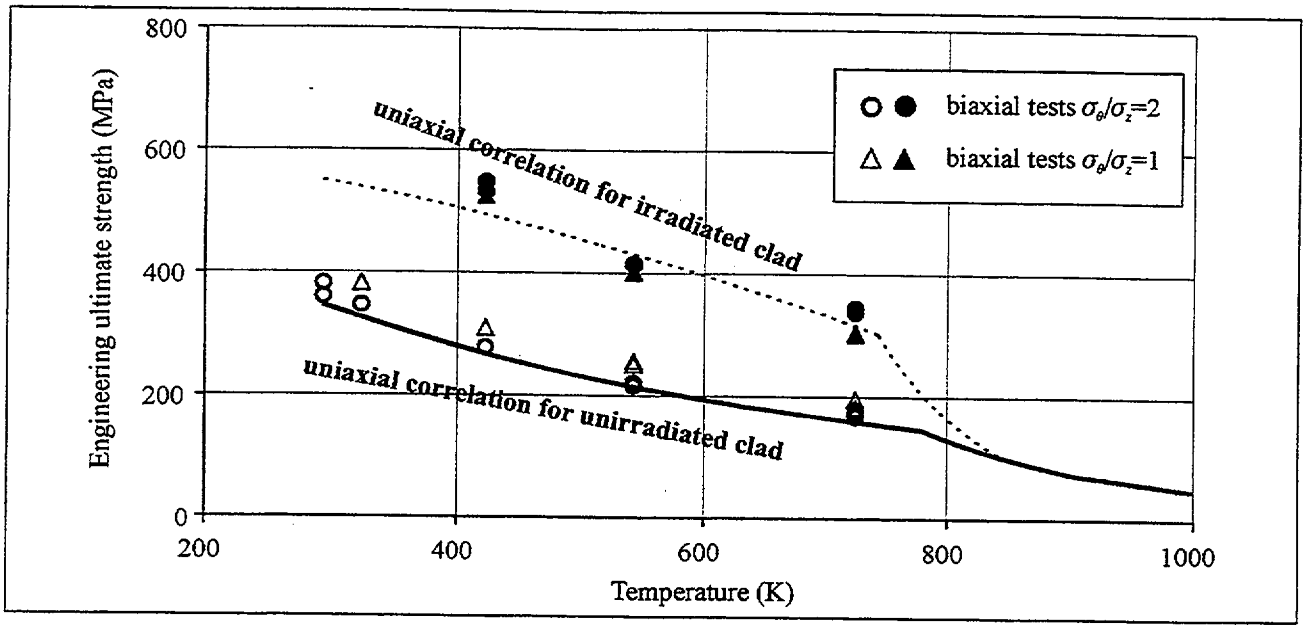

Fig. 8. Comparison of engineering ultimate strength obtained from uniaxial tensile tests and biaxial burst tests.

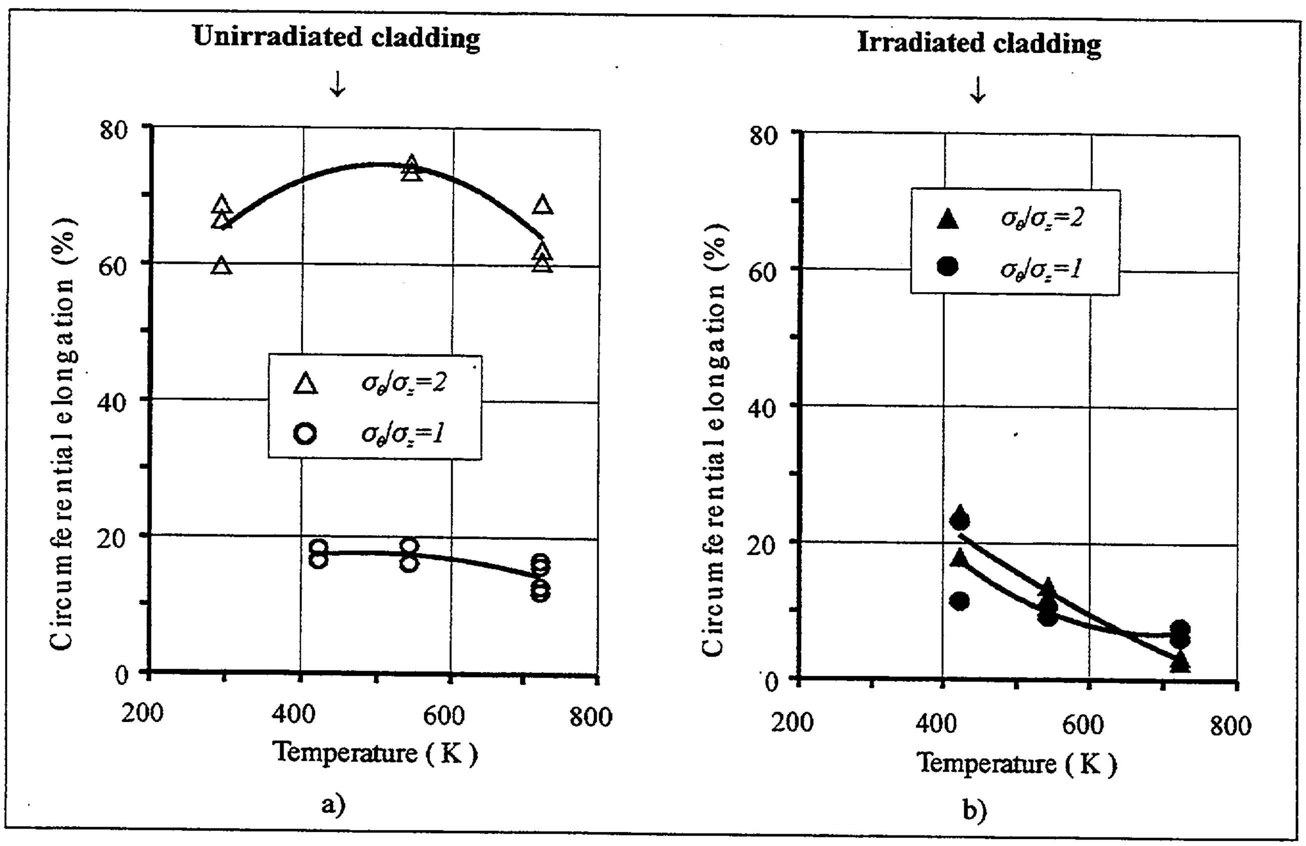

Fig. 9. Circumferential elongation at burst vs. temperature.

Failure criterion for PCMI

On the basis of uniaxial and biaxial data obtained within the temperature range 293 – 723 K, the attempt was made to propose the cladding failure criterion for the PCMI stage. Rather isothermal and constant strain-rate loading of the cladding by expanding fuel allows to use the strain energy density (SED) approach which lies in the basis of the proposed cladding integrity model [4]. Sufficient ductility of irradiated Zr-1%Nb cladding that is negligibly corroded without any local extremes and quite classical elasto-plastic stress-strain curves at uniaxial tension are also the important reasons to apply the SED approach. However, the problem arises to found the critical SED value, which is determined by integrating the stress-strain curve. Namely, the question "what the upper limit of the integral is appropriate for the PCMI case?" is now under discussion. The problem concerns with certain conservatism of uniform elongation (UE) limit and, on the other hand, with the senselessness of stress-strain curve portion from the UE to total elongation (TE). The mentioned senselessness is the inherent feature of the necking materials that fail after the significant plastic deformation localizes in a generally unknown part of the tested specimen. As for uniform elongation, it is determined fairly correctly during uniaxial testing. The conservatism of UE limit directly depends on the additional strain that cladding can accommodate without a failure after reaching the UE, i.e., the point of maximum engineering load. Therefore, special experimental and analytical efforts are evidently required to clarify the critical elongation of the real cladding subjected to deformation driven by fuel expansion. Before the results of such efforts are available, the critical SED based on UE limit can be proposed for the ductile cladding as a PCMI failure criterion. Fig. 10 presents the proposed criteria obtained from the uniaxial tensile data both for unirradiated and for irradiated cladding.

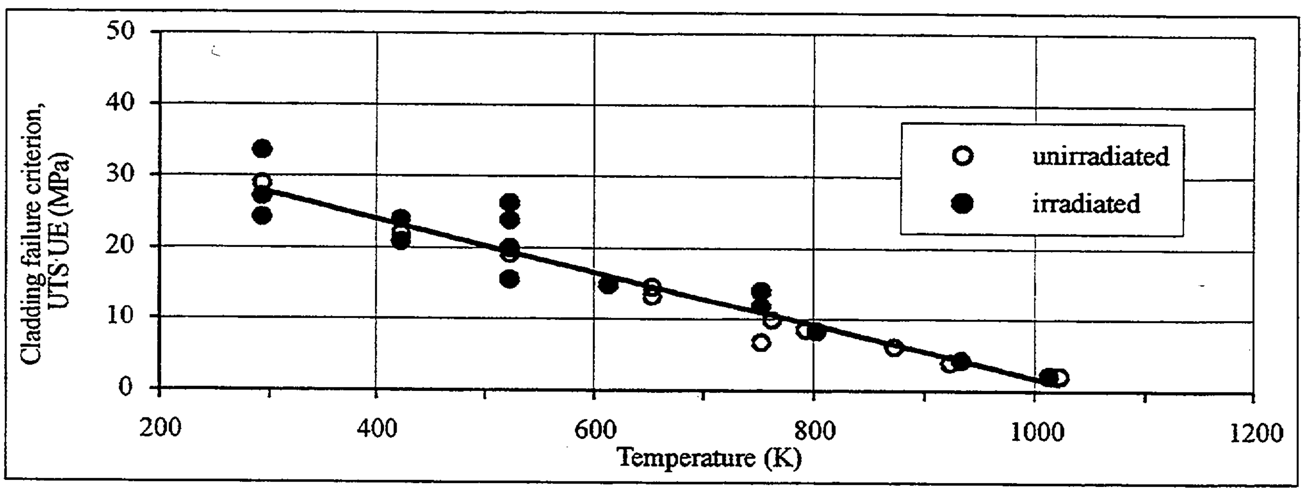

Fig. 10. Strain energy density determined up to uniform elongation for Zr-1%Nb cladding.

Conclusions

1. During 1997–1999, a special research program was performed to study the mechanical properties of unirradiated and irradiated Zr-1%Nb cladding under accident conditions.

2. The main goal of this program was to reveal the specific features of mechanical behavior of irradiated cladding and to develop the failure criteria and data base on mechanical properties of Zr-1%Nb cladding versus key parameters.
3. A set of different types of out-of-pile mechanical tests was performed and corresponding results were obtained.
4. The analytical studies concerning the computer codes problems were also performed.
5. The obtained experimental and analytical results showed that:

- data from the uniaxial transverse tensile test can be used as the major constituent of the data base with mechanical properties for the accident analysis;
- there is practically no differences between mechanical properties of unirradiated and irradiated claddings at the temperature above 900 K;
- anisotropy of mechanical properties influences significantly the parameters of deformation of unirradiated Zr-1%Nb cladding;
- further efforts are needed to study the anisotropy of mechanical properties of irradiated cladding at low temperature;
- failure criteria for unirradiated and irradiated cladding were proposed for PCMI and HTRC stages of accidents.

References

1. L.Yegorova, V.Asmolov, G.Abyshov, V.Malofeev, A.Avvakumov, E.Kaplar, K.Lioutov, A.Shestopalov, A.Bortash, L.Maiorov, K.Mikitiouk, V.Polvanov, V.Smirnov, A.Goryachev, V.Prokhorov, and A.Vurim. "Data Base on the Behaviour of High Burnup Fuel Rods with Zr-1%Nb Cladding and UO_2 Fuel (VVER Type) under Reactivity Accident Conditions", NUREG/IA-0156, Vol.1, 2, 1999.
2. A.Shestopalov, K.Lioutov, L.Yegorova, G.Abyshov, K.Mikitiouk, Modification of US NRC's FRAP-T6 Fuel Rod Transient Code for High Burnup VVER Fuel, NUREG/IA-0164, 1999.
3. K.Mikitiouk, A.Shestopalov, K.Lioutov, L.Yegorova, G.Abyshov "Modification of IPSN's SCANAIR Fuel Rod Transient Code for High Burnup VVER Fuel", NUREG/IA-0165, 1999.
4. R.O.Montgomery, Y.R.Rashid, O.Ozer, and R.L.Yang "Assessment of RIA-Simulation Experiments on Intermediate and High-Burnup Test Rods", Nuclear Safety, Vol.37, No.4, 1996.

Intercomparison of Results for a PWR Rod Ejection Accident

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ABSTRACT

This study is part of an overall program to understand the uncertainty in best-estimate calculations of the local fuel enthalpy during the rod ejection accident. Local fuel enthalpy is used as the acceptance criterion for this design-basis event and can also be used to estimate fuel damage for the purpose of determining radiological consequences. The study used results from neutron kinetics models in PARCS, BARS, and CRONOS2, codes developed in the United States, the Russian Federation, and France, respectively. Since BARS uses a heterogeneous representation of the fuel assembly as opposed to the homogeneous representations in PARCS and CRONOS, the effect of the intercomparison was primarily to compare different intra-assembly models. Quantitative comparisons for core power, reactivity, assembly fuel enthalpy and pin power were carried out. In general the agreement between methods was very good providing additional confidence in the codes and providing a starting point for a quantitative assessment of the uncertainty in calculated fuel enthalpy using best-estimate methods.

Introduction

The rod ejection accident (REA) is the design-basis reactivity initiated event for a pressurized water reactor (PWR). The acceptance criterion for unacceptable fuel damage during the event is in terms of peak local fuel enthalpy. The criterion for determining if any fuel damage has occurred, for the sake of assessing the radiological consequences, has traditionally been reaching the limiting departure-from-nucleate-boiling ratio but recent research has led to the conclusion that the criterion for moderate to high burnup fuel should also be a limiting value for local fuel enthalpy [1]. When the calculation of fuel enthalpy is done with best-estimate methods it is important to also understand the uncertainty in the results. To improve our understanding of that uncertainty, an intercomparison of the results for an REA using different methods from the U.S., France, and the Russian Federation (R.F.) has been carried out. Since some of the methods treat the fuel assembly as a homogenous region, and some use an explicit representation of each fuel pin or fuel cell, the difference in results was, to a large extent, a reflection of the uncertainty introduced by the simpler representation. Hence, this intercomparison partially satisfies the overall objective of determining the uncertainty in these calculations. In this paper the methods used are discussed first and then the REA problem used for the intercomparison is specified. Results of the intercomparison are then given and conclusions, based on those results, are explained.

Calculational Methodology

The calculations carried out in the U.S. were done with the PARCS/RELAP5 code system [2] used by the U.S. Nuclear Regulatory Commission and others, those done in the R.F. were done with BARS/RELAP5 [3] which was developed at the Russian Research Centre, Kurchatov Institute, and those done in France used a code system (SAPHYR) used by the Institute for Nuclear Safety and Protection, composed of three codes, APOLLO2 [4], CRONOS2 [5], and FLICA4 [6]. The neutron kinetics part of these code systems is PARCS, BARS, and CRONOS2, respectively.

PARCS uses a nodal approximation wherein each fuel assembly is homogenized and the two-group flux is calculated in each axial mesh of either the full assembly or a subregion of the assembly. The assembly is homogenized in the sense that the cross sections are uniform across the assembly and the thermal-hydraulic parameters are calculated for an average channel representing the assembly. The power in individual fuel rods at each time of interest can be obtained from a flux reconstruction option. This latter model requires as input the pin-by-pin flux distribution for an isolated assembly at nominal conditions.

BARS uses a Green's function approach wherein each fuel pin is represented explicitly in the radial plane. The Green's functions are based on diffusion theory and the calculation is typically done using four or five energy groups. In the axial direction an harmonic expansion is used to represent the flux. Although each pin is represented explicitly in the neutronics calculation, the fuel temperature for each pin is based on an assembly-average calculation.

CRONOS2 uses two neutron energy groups and three-dimensional diffusion theory and can either represent an homogenized assembly [CRONOS2(HOM)] or represent every fuel cell explicitly in the assembly [CRONOS2(HET)]. In the latter case, cross sections are needed for homogenized fuel cells, and for cells containing a guide tube and/or a burnable poison rod or control rod, depending on assembly design. As with all the other codes, the thermal-hydraulics is obtained for the assembly average (although it does have the ability to do four thermal-hydraulic channels in an average assembly or a subchannel calculation in selected regions).

The major difference between the codes was the intra-assembly representation and the intercomparison was expected to highlight the effect of that modeling for the dynamics of the REA. However, there are other differences between the codes and it is difficult to assess their impact. For example, the codes use different data sources. PARCS and CRONOS2(HOM) use the same two-group cross sections generated with the CASMO-3 code. BARS uses four-group lambda matrices generated with the TRIFON code. Both CASMO-3 and TRIFON use the same ENDF/B libraries.

The reactor model was based on Three Mile Island Unit 1 and made use of data available for an international benchmark exercise [7]. The reactor has one-eighth core symmetry and at hot zero power (HZP) has control banks inserted as shown on Figure 1. Figure 1 also shows the burnup for each assembly which extends up to 58 GWd/t, close to the licensing limit for PWRs in the U.S.

FIGURE 1. ONE-EIGHTH CORE REPRESENTATION WITH ASSEMBLY BURNUP, GWD/T

53 A Bank 7	30 (9H)	56	31	50 B Bank 7	28	54 Bank 6	56
	58	31	55	30	54 Bank 5	26	49
		58 Bank 6	30	54	28	23	47
			50 Bank 5	29	53	41	
				49 C Bank 7	24	41	
					37		

Regulating Bank

An attempt was made to make each reactor model as equivalent as possible. All models used an axial mesh of 24 nodes within the core and the codes run with an homogenized assembly representation [PARCS and CRONOS2(HOM)] used a 2x2 mesh radially within the assembly. PARCS and BARS used the same thermal-hydraulics model (RELAP5) and all methods used similar thermal-hydraulic options to the extent possible, e.g., the same gap conductance and the same correlation for Doppler temperature in terms of pellet centerline and surface temperature. The most difficult representation for BARS to make equivalent was the core-reflector interface.

In order to assure that the models were equivalent a set of steady state calculations were defined as given in Table 1. Suggested acceptance criteria for the difference between calculations were agreed to by the participants. These are also given in the table. The results for the reactivity parameters are given in Table 2 (refer to Figure 1 for location of control rods). Also given in Table 2 are the differences relative to PARCS which was taken as the base case for convenience. The results using SAPHYR, in this table and throughout this paper, are based on using CRONOS2(HOM) and since these calculations used the same two-group data set as used by PARCS, it is not surprising that the results are almost identical. Although BARS uses different data and different neutron balance equations, the results are in good agreement with those from PARCS and CRONOS2 indicating the desired equivalence of the models in all three codes. This equivalence is further demonstrated by the comparisons of average radial and axial power distributions shown in Figures 2 and 3, respectively.

TABLE 1. COMPARISON OF STEADY STATE PARAMETERS

PARAMETER	SUGGESTED CRITERIA
k_{eff} at HZP (Banks 5-7 in)	k_{eff} should be within ±1000 pcm
Worth of Bank 5, 6, 7	Relative difference of each bank worth within ±15%
Worth of each rod in Bank 7	Absolute difference of worths within (not specified)
Doppler defect (all rods out)	Relative difference within ± 25%
Isothermal temperature coefficient (HZP with 5 K change)	Difference in ITC within ±4 pcm/K
Radial core power distribution at HZP	Relative difference for individual assembly within ± 10%

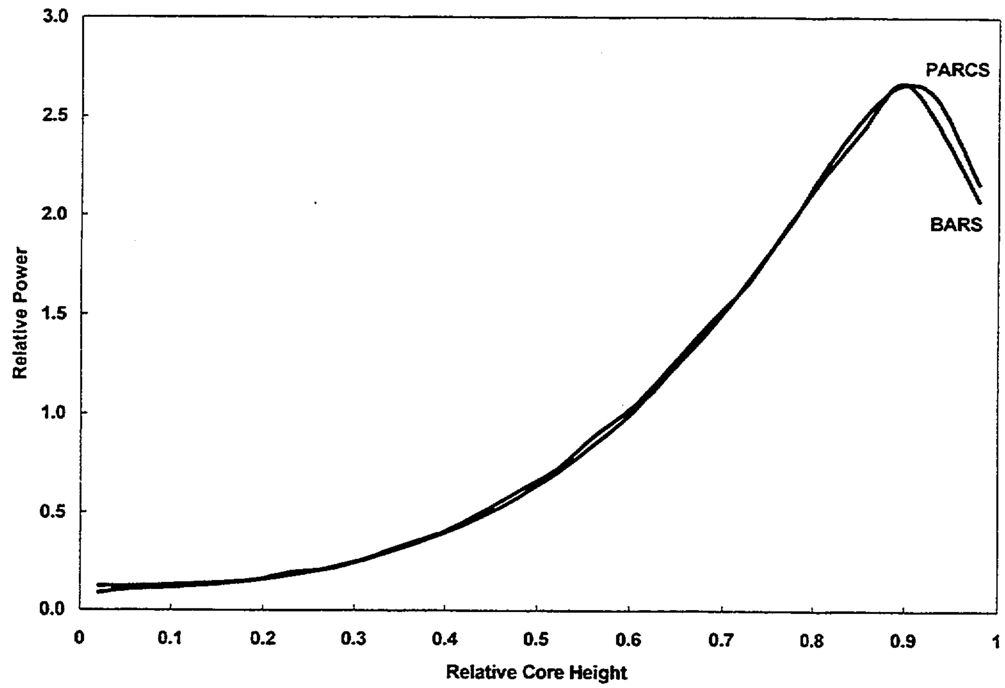
TABLE 2. STEADY STATE PARAMETERS			
	PARCS/R5	BARS/R5 (B-P)	SAPHYR (S-P)
k-eff	1.00187	1.00226 (39 pcm)	1.00165 (-22 pcm)
ρ_5	1423 pcm	1524 pcm (7.1%)	1427 pcm (0.3%)
ρ_6	849 pcm	868 pcm (2.2%)	847 pcm (-0.2%)
ρ_7	1050 pcm	1085 pcm (3.3%)	1048 pcm (-0.2%)
ρ_{7A}	347 pcm	372 pcm (25 pcm)	345 pcm (-2 pcm)
ρ_{7B}	188 pcm	208 pcm (20 pcm)	188 pcm (0)
ρ_{7C}	344 pcm	473 pcm (129 pcm)	353 pcm (9 pcm)
ΔD	2382 pcm	2235 pcm (-6.2%)	2365 pcm (-0.7%)
ITC	-47.9 pcm/K	-46.8 pcm/K (1.1 pcm/K)	-47.5 pcm/K (0.4 pcm/K)

FIGURE 2. AVERAGE RADIAL POWER DISTRIBUTION, TIME=0

0.92 0.87	1.88 1.79	1.58 1.52	1.60 1.56	0.67 0.65	0.85 0.82	0.39 0.38	0.28 0.27
	1.52 1.45	1.72 1.66	1.28 1.25	1.25 1.25	0.61 0.62	0.93 0.96	0.39 0.39
		0.82 0.79	1.32 1.31	1.10 1.17	1.31 1.37	1.11 1.16	0.40 0.41
			0.68 0.67	1.24 1.26	1.05 1.08	0.79 0.82	
				0.73 0.71	1.14 1.16	0.54 0.54	
					0.66 0.67		

PARCS OR CRONOS BARS

FIGURE 3. AVERAGE AXIAL POWER DISTRIBUTION, TIME = 0



Results for the REA

The REA was defined for the center control rod at hot zero power conditions with an ejection time of 100 ms. Since that rod has an actual worth of 347 pcm and the delayed neutron fraction (β) for the core is 0.005211 this would result in a power excursion below prompt-critical. Since the only power excursions that might lead to fuel damage are above prompt critical, the control rod worth was adjusted to obtain a worth of 1.2 β ($1.2). In this way the intercomparison would test the codes for a case simulating the limiting situation. The fixed rod worth further assured that the results of the intercomparison would be dependent on the solution of the neutron kinetics equations during the event rather than on the initial conditions. The disadvantage of this approach was that each code did the modification to achieve the same rod worth differently. PARCS used a multiplier on the absorption cross section of the central assembly, BARS multiplied diagonal elements of the lambda matrix for the control rod, and CRONOS2 normalized the source term (fission rate) across the core.

The results for the total core power are given in Figure 4 for PARCS, CRONOS2, and BARS and the results for reactivity components from PARCS are given in Figure 5 along with the total reactivity from CRONOS. The behavior is typical for an REA; a power spike turned around by Doppler feedback, followed by a period of low power decaying away. In this calculation the reactor trip is delayed to simplify the intercomparison. Normally the shutdown banks would enter the core and their effect would be felt starting at around two seconds. As seen in Figure 5, the moderator feedback is as strong as the Doppler feedback but it enters delayed and, therefore, has a much smaller effect.

FIGURE 4. CORE POWER DURING REA

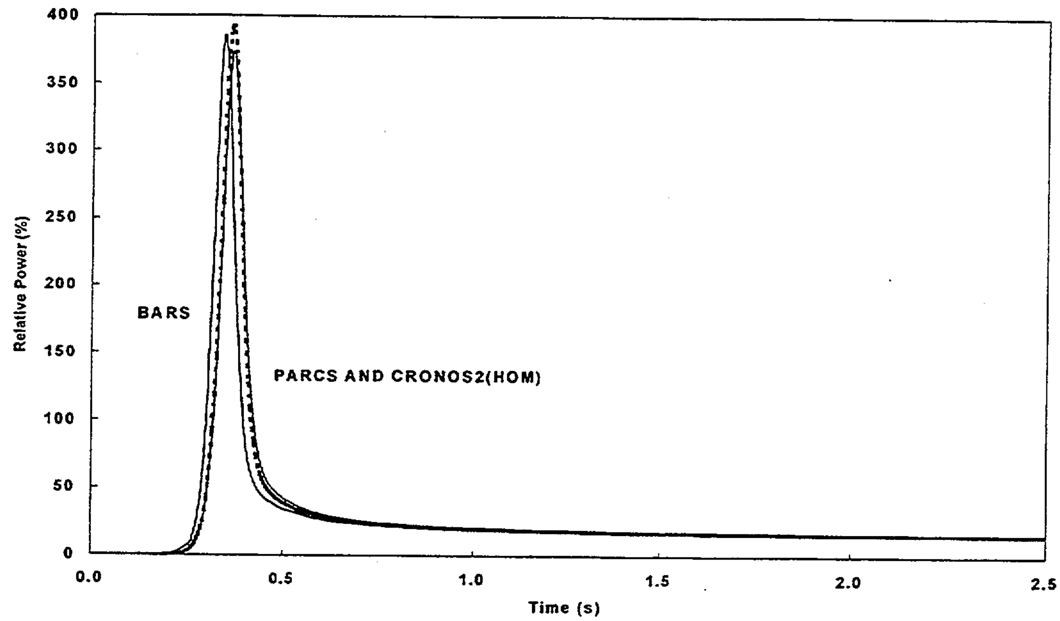
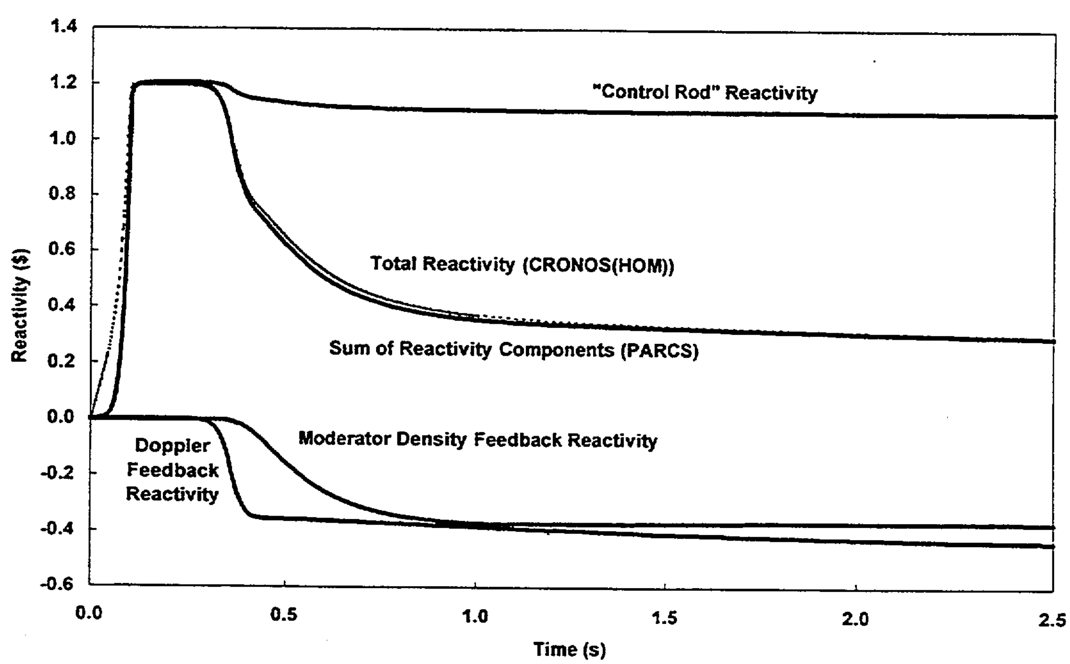


FIGURE 5. REACTIVITY COMPONENTS DURING REA



The details of the power pulse, namely, the peak power, time of the peak, and the pulse width are given for all three codes in Table 3. The agreement is very good between BARS and the two codes using the homogeneous assembly representation. Also given in Table 3 is the maximum assembly fuel enthalpy in cal/g. In all calculations this is found in axial node 22 (out of 24) at the top of the core and in Assembly 9H which is the assembly adjacent to the center assembly (see Figure 1). Note that having the peak at the top of the core is consistent with the axial power distribution shown in Figure 3. The axial power distribution is not significantly different after ejection of the central rod.

TABLE 3. TRANSIENT PARAMETERS

	PARCS/R5	BARS/R5	SAPHYR
Control Rod Worth, \$ or "β"	1.206	1.209	1.196
Peak Power, % Nominal	393	386	374
Time of Peak, ms	360	338	360
Pulse Width, ms	65	63	69
Maximum Assembly Fuel Enthalpy, cal/g	32.9	34.9	N/A

The assembly fuel enthalpy as a function of time for Assembly 9H is given in Figure 6 for both PARCS and BARS. The difference at time zero is artificial as both calculations correspond to the same initial temperature (278° C) but each code defined zero enthalpy at a slightly different temperature. The peak occurs at the end of the transient calculation (2.5 s in this case) since reactor trip has not yet been felt and the power generation is still high enough to continue to increase fuel temperature. Assuming that both calculations use the same zero point for enthalpy, the difference between the two results is approximately 2 cal/g while the increase in fuel enthalpy is approximately 18 cal/g.

The difference in assembly enthalpy at other locations is expected to be proportionally less as the increase in enthalpy at other locations is less. One indication of this is that the average assembly power distribution is tracked similarly by all three codes. Figure 7 shows the radial power at the time of the peak of the power pulse and, with the exception of the center assembly, the agreement between the results is excellent. As mentioned above, it was the center assembly that was treated differently by each code in order to obtain equal rod worths.

In this transient it is of interest to note that although the peak power (enthalpy) is located in the four assemblies adjacent to the center assembly where the rod was ejected, the power is almost as high in the 20 assemblies surrounding those assemblies. These correspond to the assemblies with burnups of 58, 56, and 31 GWd/t near the central region as shown in the one-eighth core in Figure 1.

FIGURE 6. ASSEMBLY FUEL ENTHALPY AT LOCATION OF MAXIMUM

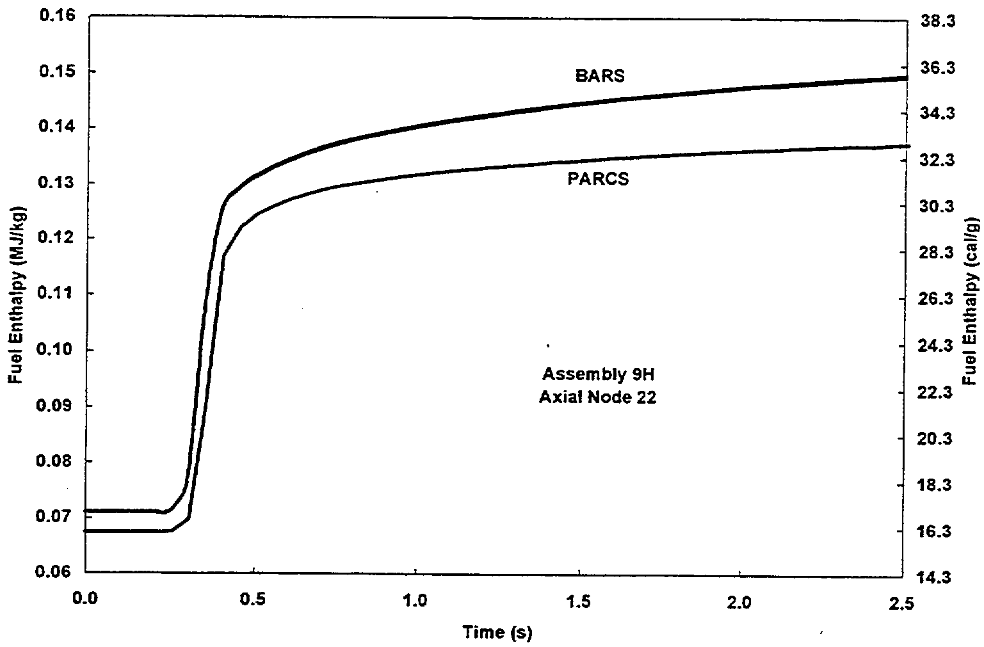


FIGURE 7. AVERAGE RADIAL POWER AT TIME = t-max

1.69 2.06 2.07	2.62 2.62 2.64	1.87 1.87 1.86	1.73 1.72 1.71	0.67 0.66 0.65	0.79 0.78 0.76	0.35 0.34 0.34	0.25 0.25 0.24
	1.97 1.97 1.95	2.00 2.00 1.98	1.36 1.36 1.34	1.23 1.22 1.22	0.57 0.56 0.56	0.82 0.82 0.83	0.34 0.37 0.33
		0.90 0.89 0.87	1.34 1.33 1.32	1.04 1.04 1.09	1.18 1.18 1.21	0.97 0.98 1.00	0.35 0.35 0.35
			0.65 0.64 0.63	1.13 1.13 1.13	0.93 0.93 0.94	0.69 0.69 0.70	
				0.64 0.64 0.62	1.00 1.00 0.99	0.46 0.46 0.46	
					0.57 0.57 0.56		

PARCS	360 ms
CRONOS	360 ms
BARS	340 ms

The parameter of most interest during the REA is the fuel enthalpy distribution (i.e., the pellet-average enthalpy for each pin at every axial mesh). Historically, this value has been obtained by taking the result for the assembly fuel enthalpy and superimposing a form function which accounts for the intra-assembly power distribution. The pin power distribution from a calculation of the isolated assembly, with reflecting boundary conditions, at nominal conditions, has sometimes been used. However, it is well-known that this is a poor way to synthesize the pin-by-pin power distribution during a transient and flux reconstruction methods exist in order to obtain results when the neutron kinetics is done with an homogenized assembly representation (e.g., as in PARCS). The flux reconstruction method provides pin power which can be integrated to obtain an estimate of fuel enthalpy or, more simply, the assembly fuel enthalpy can have the pin-by-pin power superimposed as the form function to provide the local fuel enthalpy. In either approximation, the local fuel temperature/enthalpy is not being calculated explicitly.

An intercomparison of PARCS and BARS for the pin-by-pin power along with the (small) differences seen for the assembly power gives an idea of what the intercomparison would be for the pin-by-pin fuel enthalpy. The pin power distribution within the assembly with the peak value of enthalpy is given at time zero in Figures 8 and 9 for the two codes. Figure 8 shows the distribution throughout the assembly with the orientation of the x-axis of the reactor noted (see also Figure 1) whereas Figure 9 shows the results for a trace across the assembly at the position of the x-axis. In both figures the positions where the pin power is zero correspond to the presence of either a control rod guide tube or instrumentation thimble instead of fuel.

FIGURE 8. PIN POWER FOR ASSEMBLY 9H AT TIME = 0

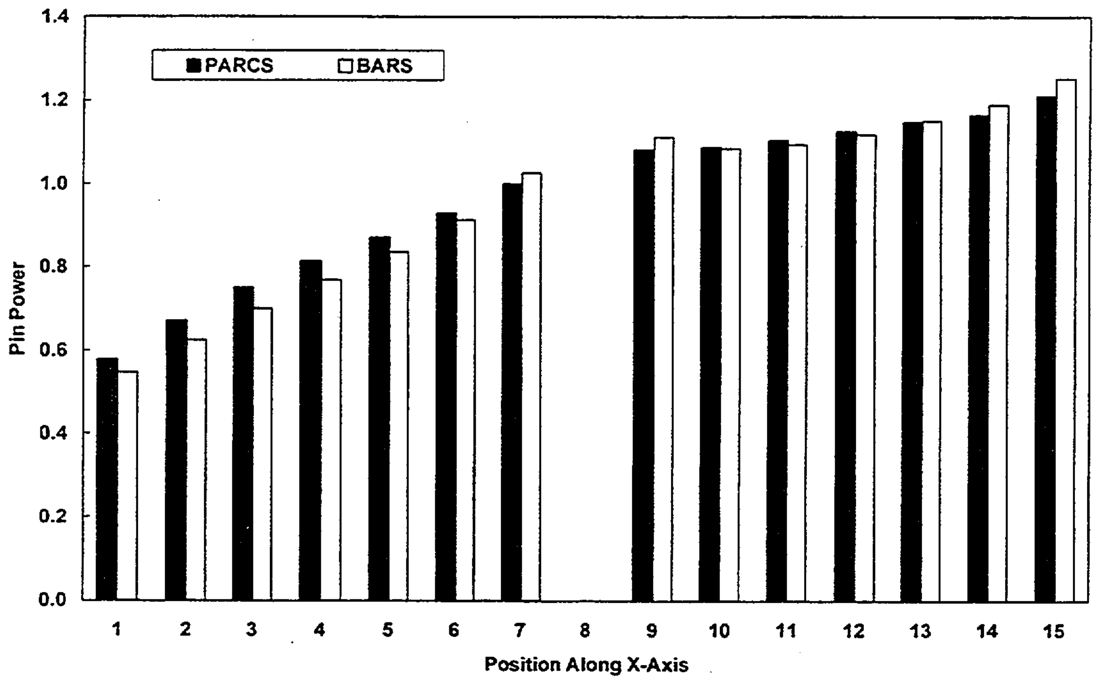
Adjacent Assembly Rodded

x-axis

0.75 0.77	0.81 0.81	0.86 0.86	0.91 0.91	0.96 0.96	1.00 1.01	1.03 1.04	1.05 1.07	1.09 1.11	1.12 1.15	1.14 1.17	1.16 1.19	1.18 1.21	1.20 1.24	1.25 1.27
0.68 0.72	0.69 0.75	0.81 0.80	0.87 0.86	0.93 0.92	0.99 0.99	1.00 1.00	1.02 1.03	1.07 1.08	1.12 1.14	1.13 1.15	1.13 1.15	1.14 1.17	1.07 1.20	1.22 1.25
0.64 0.66	0.72 0.71	0.80 0.78	0.89 0.86	0.95 0.93	0.00 0.00	1.02 1.00	1.02 1.01	1.09 1.08	0.00 0.00	1.17 1.17	1.17 1.18	1.16 1.17	1.16 1.19	1.21 1.25
0.61 0.62	0.70 0.69	0.81 0.78	0.00 0.00	0.95 0.92	0.98 0.96	0.99 0.97	1.01 0.99	1.06 1.05	1.13 1.13	1.18 1.18	0.00 0.00	1.20 1.20	1.17 1.20	1.21 1.25
0.60 0.59	0.70 0.67	0.81 0.77	0.88 0.84	0.92 0.89	0.98 0.95	0.99 0.96	1.01 0.99	1.06 1.05	1.13 1.12	1.16 1.15	1.20 1.20	1.21 1.22	1.19 1.22	1.22 1.26
0.58 0.57	0.70 0.67	0.00 0.00	0.86 0.82	0.92 0.89	0.00 0.00	1.01 0.99	1.02 1.01	1.09 1.08	0.00 0.00	1.16 1.15	1.19 1.18	0.00 0.00	1.21 1.24	1.22 1.27
0.58 0.55	0.68 0.64	0.78 0.73	0.83 0.78	0.89 0.85	0.96 0.94	1.00 1.00	1.04 1.07	1.08 1.09	1.12 1.11	1.12 1.11	1.15 1.14	1.19 1.19	1.18 1.21	1.22 1.26
0.58 0.55	0.67 0.62	0.75 0.70	0.81 0.77	0.87 0.84	0.93 0.91	1.00 1.03	0.00 0.00	1.08 1.11	1.09 1.09	1.10 1.09	1.13 1.12	1.15 1.15	1.17 1.19	1.21 1.25
0.58 0.55	0.68 0.64	0.78 0.73	0.83 0.78	0.89 0.85	0.96 0.94	1.00 1.00	1.04 1.07	1.08 1.09	1.12 1.11	1.12 1.11	1.15 1.14	1.19 1.19	1.18 1.21	1.22 1.26
0.58 0.57	0.70 0.67	0.00 0.00	0.86 0.82	0.92 0.89	0.00 0.00	1.01 0.99	1.02 1.01	1.09 1.08	0.00 0.00	1.16 1.15	1.19 1.18	0.00 0.00	1.21 1.24	1.22 1.27
0.60 0.59	0.70 0.67	0.81 0.77	0.88 0.84	0.92 0.89	0.98 0.95	0.99 0.96	1.01 0.99	1.06 1.05	1.13 1.12	1.16 1.15	1.20 1.20	1.21 1.22	1.19 1.22	1.22 1.26
0.61 0.62	0.70 0.69	0.81 0.78	0.00 0.00	0.95 0.92	0.98 0.96	0.99 0.97	1.01 0.99	1.06 1.05	1.13 1.13	1.18 1.18	0.00 0.00	1.20 1.20	1.17 1.20	1.21 1.25
0.64 0.66	0.72 0.71	0.80 0.78	0.89 0.86	0.95 0.93	0.00 0.00	1.02 1.00	1.02 1.01	1.09 1.08	0.00 0.00	1.17 1.17	1.17 1.18	1.16 1.17	1.16 1.19	1.21 1.25
0.68 0.72	0.69 0.75	0.81 0.80	0.87 0.86	0.93 0.93	0.99 0.99	1.00 1.00	1.02 1.03	1.07 1.08	1.12 1.14	1.13 1.15	1.13 1.15	1.14 1.17	1.07 1.20	1.22 1.25
0.75 0.77	0.81 0.81	0.86 0.86	0.91 0.91	0.96 0.96	1.00 1.01	1.03 1.04	1.05 1.07	1.09 1.11	1.12 1.15	1.14 1.17	1.16 1.19	1.18 1.21	1.20 1.24	1.25 1.27

PARCS	t=0
BARS	t=0

FIGURE 9. RELATIVE PIN POWER ACROSS ASSEMBLY 9H TIME = 0



For the comparison shown, the full capability of the PARCS reconstruction method was not used. Corner point discontinuity factors can be used in the reconstruction but were not available from the isolated assembly calculation and values of 1.0 were assumed. The two-group form functions were also not available and hence the pin-by-pin power distribution from the isolated assembly was used for those form functions. With these limitations the rms (root mean square) difference between the two results is 3.1% and the maximum difference (with one exception) is less than 5%, indicating good agreement between the totally different methods for treating the intra-assembly neutron kinetics. Note that this agreement is for an assembly with a steep gradient along the x-axis. As can be seen in Figure 9, the pin power increases by a factor of two across the assembly [as a result of the presence of the control rod in the (core center) assembly adjacent to the assembly being analyzed]. The exception to the good agreement is an error of ~10% in pins containing gadolinium. This is probably the result of differences in the treatment of the depletion of Gd in the codes providing data to PARCS and BARS.

A similar comparison of pin power distributions is given in Figures 10 and 11 at the time at which the total power is a maximum. In this case the power distribution across the assembly is relatively flat--all adjacent assemblies are uncontrolled--and the rms difference between the two sets of data is reduced to 1.7% and the maximum difference to less than 2% except in the gadolinium-containing pins.

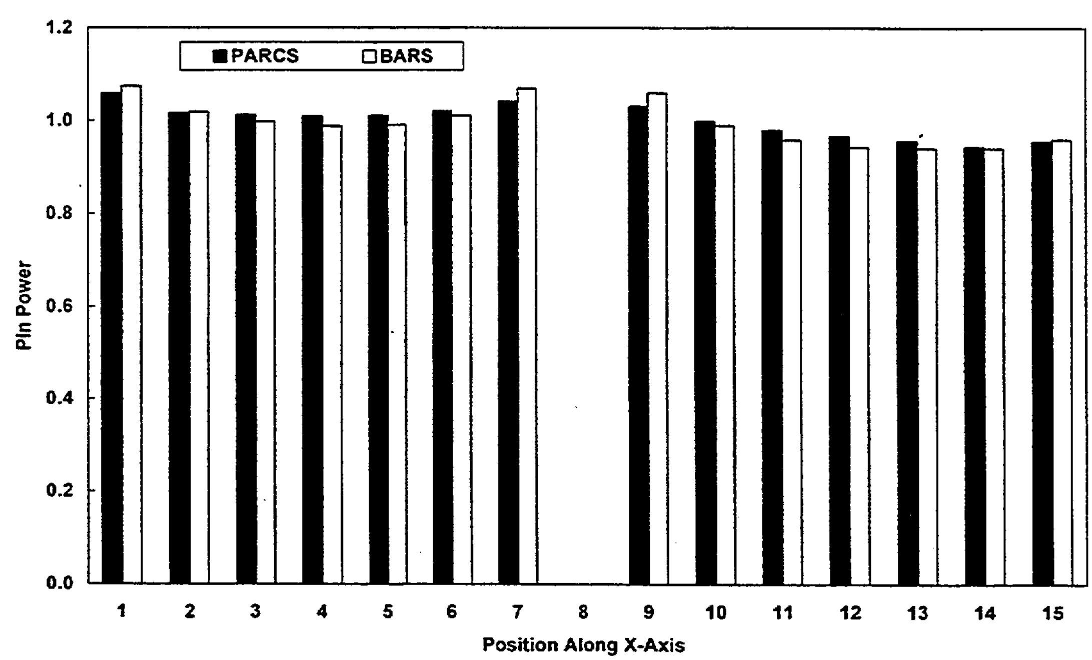
FIGURE 10. PIN POWER FOR ASSEMBLY 9H AT TIME = t-max

Adjacent Assembly Unrodded (left side); x-axis →

1.06 1.06	1.02 1.03	1.01 1.02	1.01 1.02	1.01 1.02	1.01 1.02	1.00 1.01	0.99 1.00	0.99 0.99	0.98 0.99	0.97 0.98	0.96 0.96	0.95 0.95	0.94 0.94	0.96 0.94
1.05 1.06	0.92 1.01	0.99 0.99	0.99 0.99	1.01 1.01	1.02 1.02	0.99 0.99	0.97 0.97	0.98 0.98	1.00 1.00	0.97 0.97	0.95 0.94	0.93 0.93	0.85 0.92	0.94 0.94
1.05 1.06	1.00 1.00	1.01 1.00	1.04 1.03	1.05 1.04	0.00 0.00	1.02 1.00	0.98 0.96	1.01 0.99	0.00 0.00	1.02 1.00	0.99 0.98	0.95 0.94	0.93 0.92	0.94 0.94
1.06 1.07	1.01 1.02	1.05 1.03	0.00 0.00	1.07 1.05	1.05 1.03	1.01 0.99	0.99 0.96	1.00 0.98	1.02 1.01	1.03 1.01	0.00 0.00	0.99 0.97	0.94 0.94	0.95 0.95
1.06 1.08	1.03 1.04	1.07 1.06	1.07 1.06	1.05 1.03	1.05 1.03	1.02 1.00	1.00 0.97	1.00 0.98	1.03 1.01	1.02 1.00	1.03 1.01	1.01 0.99	0.96 0.96	0.96 0.96
1.07 1.08	1.06 1.06	0.00 0.00	1.06 1.04	1.06 1.04	0.00 0.00	1.04 1.03	1.01 1.00	1.03 1.02	0.00 0.00	1.02 1.00	1.01 1.00	0.00 0.00	0.98 0.98	0.96 0.97
1.06 1.08	1.03 1.03	1.05 1.03	1.03 1.01	1.03 1.01	1.05 1.04	1.04 1.04	1.04 1.07	1.03 1.03	1.03 1.01	0.99 0.97	0.98 0.96	0.99 0.97	0.96 0.95	0.96 0.96
1.06 1.07	1.02 1.02	1.01 1.00	1.01 0.99	1.01 0.99	1.02 1.01	1.04 1.07	0.00 0.00	1.03 1.06	1.00 0.99	0.98 0.96	0.97 0.94	0.96 0.94	0.94 0.94	0.96 0.96
1.06 1.08	1.03 1.03	1.05 1.03	1.03 1.01	1.03 1.01	1.05 1.04	1.04 1.04	1.04 1.07	1.03 1.03	1.03 1.01	0.99 0.97	0.98 0.96	0.99 0.97	0.96 0.95	0.96 0.96
1.07 1.08	1.06 1.06	0.00 0.00	1.06 1.04	1.06 1.04	0.00 0.00	1.04 1.03	1.01 1.00	1.03 1.02	0.00 0.00	1.02 1.00	1.01 1.00	0.00 0.00	0.98 0.98	0.96 0.97
1.06 1.08	1.03 1.04	1.07 1.06	1.07 1.06	1.05 1.03	1.05 1.03	1.02 1.00	1.00 0.97	1.00 0.98	1.03 1.01	1.02 1.00	1.03 1.01	1.01 0.99	0.96 0.96	0.96 0.96
1.06 1.07	1.01 1.02	1.05 1.03	0.00 0.00	1.07 1.05	1.05 1.03	1.01 0.99	0.99 0.96	1.00 0.98	1.02 1.01	1.03 1.01	0.00 0.00	0.99 0.97	0.94 0.94	0.95 0.95
1.05 1.06	1.00 1.00	1.01 1.00	1.04 1.03	1.05 1.04	0.00 0.00	1.02 1.00	0.98 0.96	1.01 0.99	0.00 0.00	1.02 1.00	0.99 0.98	0.95 0.94	0.93 0.92	0.94 0.94
1.05 1.06	0.92 1.01	0.99 0.99	0.99 0.99	1.01 1.01	1.02 1.02	0.99 0.99	0.97 0.97	0.98 0.98	1.00 1.00	0.97 0.97	0.95 0.94	0.93 0.93	0.85 0.92	0.94 0.94
1.06 1.06	1.02 1.03	1.01 1.02	1.01 1.02	1.01 1.02	1.01 1.02	1.00 1.01	0.99 1.00	0.99 0.99	0.98 0.99	0.97 0.98	0.96 0.96	0.95 0.95	0.94 0.94	0.96 0.94

PARCS	t=t-max
BARS	t=t-max

FIGURE 11. RELATIVE PIN POWER ACROSS ASSEMBLY 9H TIME = t-max



Summary

An intercomparison of results for an REA has been carried out using three different code systems used in the U.S., the R.F., and France. The neutron kinetics in these systems is based on the PARCS, BARS, and CRONOS2 codes, respectively. The results give an indication of the uncertainty in the calculated local fuel enthalpy. In particular, since two very different methods of solving the space-dependence of the flux across an assembly, the intercomparison gives an indication of the uncertainty that arises due to the spatial representation within the fuel assembly. PARCS and CRONOS use an homogenized assembly representation whereas BARS uses a heterogeneous representation of each fuel pin. In the future an additional comparison will be made with a version of CRONOS that models each fuel cell explicitly.

In order to limit the differences between codes to the intra-assembly representation to the extent possible, an attempt was made to make the models equivalent in all other ways. The most difficult aspect of this was the treatment of the core-reflector interface. The success of the equivalence was judged by the good agreement obtained for steady state power distribution and control rod and thermal-hydraulic feedback reactivity. All quantities were within limits established by the participants prior to the start of calculations.

Several quantities were compared during the transient. Total power, including the peak power, time of the peak, and pulse width were in excellent agreement using all three methods. The calculated assembly average fuel enthalpy at the axial location at which it was greatest during the REA increased by approximately 18 cal/g from an initial value of 14 cal/g. The difference between the peak assembly fuel enthalpy using an homogenized assembly (PARCS and CRONOS) and using a heterogeneous model which represented each fuel pin explicitly (BARS), was 2 cal/g. The differences are expected to increase with an increase in the spatial gradient of power. The expected variation can be tested with the data generated to date by looking at the results for different assemblies at different axial locations. The results also demonstrated that a large region of the core can be affected by an REA.

The calculated pin power was also compared using the flux reconstruction model in PARCS and the BARS heterogeneous model. The results showed excellent agreement except for fuel pins containing gadolinium with the rms difference at time zero being only 3.1% and at the time of the peak power the rms difference was 1.7%. The difference between the two models was expected to increase with increasing gradient and these results are consistent with that expectation. The power across the assembly varied by more than a factor of two at time zero whereas at the time of the peak power the variation was only ±8%.

In the future it is expected that additional intercomparisons will be available with the heterogeneous version of CRONOS2 and that additional assemblies will be considered so that the uncertainty due to the spatial representation can be assessed more definitively. That work in conjunction with other efforts to understand the uncertainty as a result of variations in Doppler feedback, control rod worth, delayed neutron fraction, etc. should lead to a better estimate of the uncertainty in calculated fuel enthalpy using best-estimate methods.

References

1. R.O. Meyer et al., "A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents," *Nuclear Safety, 37*, No. 4, October-December 1996.

2. H. G. Joo et al., "PARCS: A Multi-Dimensional Two-Group Reactor Kinetics Code Based on the Nonlinear Analytic Nodal Method," PU/NE-98-26, Purdue University, September 1998.

3. A. Avvakumov and V. Malofeev, "Validation of an Advanced Heterogeneous Model for LWR Detailed Pin-by-Pin Calculations," Proceedings of the International Conference on the Physics of Nuclear Science and Technology, Long Island, NY, October 1998.

4. S. Loubière et al., "APOLLO2 Twelve Years Later," NT SERMA/LENR/PU/99-2557/B

5. J. J. Lautard, S. Loubière, C. Fedon-Magnaud, "CRONOS2, A Modular Computational System for Neutronic Core Calculations," IAEA Specialist Meeting on Advanced Calculational Methods for Power Reactors, Cadarache, France.

6. I. Toumi, D. Gallo, A. Bergeron, and E. Royer, "Advanced Numerical Methods for Three-Dimensional Two-Phase Flow Calculations in PWR," NURETH-8 Conference, Kyoto, Japan.

7. K.N. Ivanov et al., "Pressurized Water Reactor Main Steam Line Break (MSLB) Benchmark; Volume I: Final Specifications," NEA/NSC/DOC(99)8, U.S. Nuclear Regulatory Commission and OECD Nuclear Energy Agency, April 1999.

Spent Fuel Burnup Credit in Casks: An NRC Perspective

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Abstract

Until now, the Nuclear Regulatory Commission's (NRC) approval of criticality safety evaluations for spent fuel in transport and storage casks has been based on analyzing the fuel as though it were fresh and without burnable poisons. The well-known nuclide composition of fresh fuel has provided a straightforward and bounding approach for showing that spent fuel systems will remain subcritical under normal and accident conditions. Burnup credit refers to the approval of criticality safety evaluations that consider the decrease in fuel reactivity caused by irradiation in the reactor. Extensive investigations have been performed in the U.S. and other countries to understand and document the technical issues related to burnup credit. This paper reviews the background for NRC's efforts toward applying burnup credit in the licensing of casks for spent fuel from pressurized water reactors, discusses technical issues affecting the evolving NRC guidance in this area, and outlines the information and efforts needed to further expand such applications of burnup credit.

Introduction

When fuel is irradiated in a reactor, the reactivity of the fuel changes. The variation of fuel reactivity with irradiation is governed by the fuel's changing composition of fissile actinides, non-fissile actinides, fission products, and internal burnable poisons. Ignoring the initial presence of burnable poisons, which may be regarded as fully depleted in spent fuel, the remaining composition changes will cause the net reactivity of the fuel to decrease. Until now, the U.S. Nuclear Regulatory Commission's (NRC) approval of the criticality safety evaluations for commercial spent fuel in casks, including storage, transport, and dual-purpose casks, has been based on analyzing the spent fuel as though it were unirradiated and without burnable poisons. This "fresh-fuel" assumption has provided a straightforward and bounding approach for showing that spent fuel packages will remain subcritical under normal and accident conditions. The extreme conservatism of the fresh-fuel assumption, however, can lead to excessive design requirements for neutron absorbers and/or spacing of the spent fuel.

The term burnup credit refers to allowing the criticality safety of spent fuel systems to be evaluated using analysis approaches that consider the reduced reactivity of irradiated fuel. Actinide-only methods of burnup credit analyze only the effects of actinides on fuel reactivity. In commercial power-reactor fuels that have achieved most of their intended burnup, actinide effects generally account for well over half of the change in reactivity relative to the fresh-fuel assumption, with fission products accounting for the remainder. In the U.S., interest in burnup credit for spent fuel casks has focused mainly on fuel from pressurized water reactors (PWRs) rather than from boiling water reactors (BWRs). This is largely because the smaller pin-array size and correspondingly lower reactivity of individual BWR assemblies, in relation to PWR assemblies, leads to relatively small economic penalties in cask design and capacity when analyzed under the fresh-fuel assumption. Another factor is that the neutronically more complex and variable operation of BWR fuels tends to substantially limit their analysis for burnup credit. This paper reviews the background for NRC's efforts toward applying burnup credit to PWR spent fuel in casks, discusses technical issues affecting the evolving NRC guidance in this area, and outlines the information and efforts needed to further expand such applications of burnup credit.

Background

Other NRC Uses of Burnup Credit in Spent Fuel Storage

The Office of Nuclear Reactor Regulation (NRR) has long allowed the use of burnup credit in the borated spent-fuel storage pools at PWR plants.[1] This is based in part on the established ability of licensees to predict the core burnup behavior over hundreds of reactor years of operation. Additional safety assurance is based on application of the double contingency principle as defined in ANSI/ANS-8.1-1983,[2] and in Title 10, Code of Federal Regulations (10 CFR), Section 72.124(a),[3] which requires two unlikely, independent, concurrent events to produce a criticality accident. For example, if soluble boron is normally present in the spent fuel pool water, the loss of soluble boron is considered as one accident condition and a second concurrent accident need not be assumed. Alternatively, credit for the presence of soluble boron in PWR pools may be assumed in evaluating other accident conditions such as the misloading of fresh fuel assemblies into racks restricted to irradiated fuel. Typically, there is sufficient soluble boron in PWR pools to maintain at least a 5% subcriticality margin even if an entire burnup-dependent storage rack were misloaded with fresh fuel assemblies.

As noted by DOE and others, burnup credit calculations can also be found in the applicants' safety analysis reports (SARs) for two NRC-approved single-purpose dry storage casks for PWR spent fuel. In those cases, the applicants performed burnup credit calculations in evaluating hypothetical underboration events during wet loading or unloading of the dry storage casks. However, the NRC staff's safety evaluation reports for those cases used the fresh-fuel analysis assumption in combination with credit for boron in the water. Boron credit was made possible by creating in the license or certificate a Technical Specification requiring two independent verification controls to ensure sufficient soluble boron concentration during wet loading and unloading operations. This satisfied the double-contingency criterion of 10 CFR 72.124(a) while obviating consideration of loss-of-boron events in the review under 10 CFR Part 72.

Consideration of burnup credit after drying and closure of casks is not necessary in 10 CFR Part 72 storage applications because it has been shown that the probability of fresh-water ingress into sealed dry storage casks is sufficiently low. Specifically, the double-contingency criterion is satisfied by showing that water ingress into a storage cask would require both a flooding event and a severe accident that would cause gross seal failure. On the other hand, transportation regulations under 10 CFR Part 71 include explicit requirements for assuming fresh-water inleakage in the criticality analysis of packages used for transporting fissile materials.[4] Sections 6.5.4 and 6.5.5 in NUREG-1617, "Standard Review Plan for

Transportation Packages for Spent Nuclear Fuel," further discuss the water-inleakage considerations for spent-fuel evaluations under 10 CFR Part 71.[5]

Burnup Credit in Other Countries

Several regulatory bodies outside the U.S. have allowed various uses of burnup credit in wet storage and handling operations, and also in reprocessing. However, transportation uses of burnup credit have been granted to-date only in France. The French reprocessing program has developed an extensive set of proprietary validation data to support the limited credit needed for shipping modern PWR fuels with higher initial enrichments in the existing fleet of casks. Safety authorities in several other countries, including the United Kingdom and Japan, are now working toward similar uses of burnup credit in transport packages. The NRC research program is now evaluating options for acquiring validation benchmarks from French and other foreign or proprietary sources as needed to support expansion of the scope and level of NRC-approved burnup credit in casks. As an indicator of the high level of interest in this field, it is noteworthy that this year's International Conference on Nuclear Criticality Safety featured four technical sessions with nineteen papers devoted to the uses of burnup credit.[6]

NRC Guidance on Burnup Credit Methods for PWR Spent Fuel in Casks

The U.S. Department of Energy (DOE) has worked on the development of a topical report that proposes a method for incorporating actinide-only burnup effects in the analysis of casks for transporting and storing spent fuel from PWRs.[7] The topical report has gone through two cycles of revisions in response to NRC's review and comments, yet outstanding technical issues and uncertainties have prevented NRC from granting the requested approval. Nevertheless, based in part on the technical information provided in the DOE topical report, and supplemented by information available from other sources, the NRC Spent Fuel Project Office (SFPO) issued in May 1999 the initial version of its interim staff guidance document, ISG-8 Revision 0.[8] That initial guidance recommended approving the DOE methodology for use only when the spent fuel is modeled at 50% of the verified and adjusted burnup level from plant records. This 50% limit on the assumed burnup served to cover the staff's remaining issues and uncertainties concerning the proposed methodology. On May 17, 1999, the NRC staff held a public workshop[9] to introduce ISG-8 and discuss NRC and industry perspectives on the further development of burnup credit for PWR spent fuel in casks.

To support the staff's phased efforts in this area, the NRC initiated a research program on burnup credit in early 1999. That research program is the topic of another paper to be presented at this meeting.[10] The initial phases of the research program have included an analysis effort focused on supporting the early revision of ISG-8 to allow greater levels burnup credit. On July 30, 1999, the first results from that effort enabled SFPO's issuance of Revision 1 of ISG-8.[11] This completely rewritten version of the NRC guidance recommends a basis for cask-specific approval of PWR actinide-only burnup credit analyzed at essentially 100% of an assembly's verified and adjusted burnup level from plant records. One of the main limitations of the guidance is that its direct application is restricted to PWR fuels that have not used burnable absorbers. It is worth noting that the same restriction is found in the method proposed by DOE. A copy of Revision 1 of ISG-8 is included herein as Appendix A. Comments on related modeling and validation issues that affect the further evolution of ISG-8 are included in bulleted form in Appendix B.

Technical Considerations for Revision 1 of ISG-8

This section briefly discusses the technical considerations for selected aspects of Revision 1 of ISG-8. In particular, these comments pertain to Items 1 and 6 of the ISG-8 recommendations.

Item 1 in the Recommendations Section of ISG-8 Rev.1 allows the use of a so-called loading offset for spent fuels with initial ^{235}U enrichments between 4 and 5%. The offset effectively reduces the burnup assumed in calculating the actinide inventories in the affected fuels. The need for this offset arises from the lack of isotopic assay data from spent fuels in the 4 to 5% enrichment range. This offset is an example of how conservative modeling adjustments can be judiciously used to compensate for validation uncertainties that arise from modest extrapolations beyond the measured data.

In establishing the adequacy of the loading offset approach within the current context, the staff has noted the following: All other factors being equal, an increase in initial enrichment lowers the contribution from actinides to the reduced reactivity of spent fuel, thereby increasing the relative contribution from fission products. Thus, the neglect of fission products in actinide-only burnup credit is especially helpful in further offsetting the uncertainties from this limited extrapolation to higher initial enrichments. Such would not be the case if one were to consider an extrapolation to higher burnups. This is because the actinide contribution to reducing the reactivity of irradiated fuel increases much more rapidly with burnup than does the contribution from fission products.

Item 6 in the Recommendations Section of ISG-8 calls for the applicant to provide design-specific analyses that estimate the additional reactivity margins available from fission product and actinide nuclides not included in the licensing safety basis. As discussed below, this recommendation arises from the staff's efforts at addressing the following question: Can the combined effects of uncertainties and approximations in actinide-only burnup credit outweigh the margins from the neglect of fission products and ^{236}U?

At three points in DOE's topical report (Sections 3.2, 4.1.5, and 4.2.3.3), a portion of the large reactivity margins arising from the method's neglect of fission products and ^{236}U is used in attempting to bring closure to an issue. In response to requests from the NRC staff, the current Revision 2 of the topical report now includes in Table 7-4 a tally of the uses of estimated fission-product (and ^{236}U) reactivity margins. Specifically, for initial enrichments of 3.0, 3.6, and 4.5 wt% ^{235}U and burnups of 15, 30, and 45 GWD/MTU, the table subtracts from the estimated fission-product margins three reactivity allowances to account for (a) the unmodeled higher reactivity of fuel assemblies in which control rods were inserted during part of the burnup and (b) uncertainties associated with criticality validation and computer code adequacy issues. The report's tabulated results show a residual margin of at least 2.1% Δk_{eff} in all cases. The topical report and its references, however, fail to provide necessary information about the assumptions and models used in estimating the fission-product and ^{236}U margins and in establishing allowances for the higher reactivity of fuels burned with control rods inserted.

The staff's initial confirmatory analyses, performed with assistance from the NRC research program, have been focused on understanding the estimation and uses of the fission-product and ^{236}U reactivity margin. The NRC's calculations on different cask models have demonstrated that the estimated fission-product and ^{236}U margins vary substantially between cask designs. For example, higher poison loadings in the basket reduce the margins by capturing neutrons otherwise absorbed by fission products. Some of the cask models analyzed by the NRC have yielded calculated fission-product and ^{236}U margins significantly smaller than those in DOE's topical report. As shown by example in Table 1, subtracting the topical report's three reactivity allowances from the NRC-calculated margins for fission products and ^{236}U was found to leave negative residual margins at certain values of low initial enrichment and low burnup. This result can be explained in part by noting that DOE's assumed reactivity allowances for the reactivity effects of burnup in the presence of control rods are greatest at low initial enrichments and constant beyond burnups of 15 GWD/MTU. It is possible, however, that such combinations of low burnup and low initial enrichment would fall below the burnup credit loading curve for the respective cask design.

In response to NRC questions, section 7.4 of the DOE topical report (i.e., Rev.2) discussed several smaller margins, in addition to those from neglecting fission products and ^{236}U, that are associated with apparent modeling conservatisms in the report's actinide-only methodology for burnup credit. Such additional margins would generally tend to offset some or all of the negative residual margins that might appear in cask-specific versions of DOE's Table 7-4. However, as noted in the topical report, the magnitudes of the individual margins are relatively small, variable, and poorly quantifiable. More importantly, most of the additional margins are based on comparisons against the typical or mean case and therefore do not cover the full range of possible or credible fuel loadings that would be allowed under the report's burnup credit method. The NRC staff therefore concludes that it is not possible, based on information considered to-date, to ensure categorically that the aggregate of such additional margins is large enough to offset actinide-only uncertainties in casks where the margins from the neglect of fission products are especially small. The staff expects that further insights into the existence and magnitude of residual margins will emerge from the applicants' cask-specific estimates of (1) the margins from neglected fission product and actinide nuclides and (2) the reactivity effects of uncertainties and potential nonconservatisms in the actinide-only methods.

What Next?

Comments received to-date have indicated an interest in extending burnup credit methods to include PWR fuels that have used burnable or removable absorbers. For work to proceed in this area, the NRC staff and the burnup-credit applicants will need information on the past and present uses of burnable or removable absorbers in PWR fuel designs.

From the preceding discussion on estimation of additional margins, it also appears that a better understanding is needed of reactivity effects in PWR fuels in which control rods were inserted during a significant portion of the burnup history. In particular, a better assessment of the scope and magnitude of rodded burnup histories in the worst-case operating cycles at worst-case PWR plants is needed to support the NRC staff's evaluation of approaches to higher levels of burnup credit.

Therefore, the NRC staff is requesting assistance from the industry in compiling the following types of information:

1. Past and present uses of removable and burnable internal poisons in PWR fuel designs:

(a) What poison materials (e.g., Gd, B, Er) have been used, in what amounts (e.g., in grams of poison per cm^3, or per cm of fuel pin, or per cm of fuel assembly), in what form (i.e., mixed with UO_2, coated on fuel pellets, in poison-only rods, etc.), and in what geometry (i.e., representative pin-by-pin poison zonings)?

(b) Information from the preceding item collated, as warranted, over ranges of (i) initial U-235 enrichment, (ii) design or actual burnup, and (iii) assembly design geometry (i.e., grouped by fuel vendor and pin array size - B&W15x15, W17x17, CE14x14, etc.).

(c) For movable or removable poisons, the typical or bounding histories of poison use in an assembly (e.g., 15 GWD/MTU, first-cycle-only, part-length axial location and extent, etc.)

2. Past and present uses of control rods for load following and power shaping in U.S. PWRs:

   The NRC staff is aware that at-power insertion of control rods for load following has not been extensively practiced in the U.S. However, because the NRC licenses cask designs to take spent fuel from many or all plants, it is important to know about the worst-case rodded burnup histories at the worst-case plants. Of interest would be an identification of the worst-case plants and cycles and, for those plants and cycles, information on what kinds of control rods were used, how deep, how long (i.e., in terms of burnup), in which burnup cycles (e.g., first only) and in how many and which kinds of assemblies.

All other things being equal, spent fuel burned in the presence of thermal neutron poisons can have a significantly higher k_{eff} than fuel burned without poisons. This is because the poisons harden the energy spectrum of neutrons absorbed by fuel, leading to more breeding of fissile Pu. While the poisons may fully deplete with burnup, the more reactive actinide compositions remain significantly intact. In considering burnup credit, the NRC and its applicants will need to determine which categories of internal poison designs bound others with respect to the computed in-package k_{eff} of spent fuel.

The higher reactivity of spent fuel burned in the presence of poisons is a strong function of initial enrichment; namely, lower-enriched fuels show a much stronger reactivity effect for a given burnup and poison loading. This is because lower-enriched fuel has less initial ^{235}U to deplete, yet breeds fissile Pu faster, per unit of burnup, than higher-enriched fuel. Therefore, the bounding poison categories may have to be evaluated over two or more ranges of initial ^{235}U enrichment and final assembly-average burnup (see item 1(b) above).

The requested information on internal poisons should be comprehensive - representing essentially all fuel cycles and assemblies at all U.S. PWRs - to enable consideration of significant burnup credit for all such fuels. To the extent that detailed information on fuel designs may be proprietary, appropriate measures will be taken to either protect the proprietary interests or else convert the information to a less-detailed, nonproprietary form that still meets the needs of NRC and its applicants.

The information on worst-case rodded burnup histories is especially relevant when considering the fact that any fission chain reactions in spent fuel happen at the less-burned top ends of the fuel assemblies. It appears that worst-case axial burnup profiles may be closely correlated with rodded burnup histories. If that is true, we must account not only for the lower burnup at the top of the fuel, but also the more reactive actinide compositions associated with part of that burnup having occurred in the presence of control rods.

The level and scope of appropriate burnup credit is proportional to the information available. The current Revision 1 of ISG-8 does not include burnup credit for PWR fuels burned with internal poisons and does not address fission product credit. The NRC staff would like to lessen those restrictions, but needs the kinds of information described above in order to proceed.

References:

1. Laurence Kopp, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," NRC Memorandum to Timothy Collins, U.S. Nuclear Regulatory Commission, August 19, 1998.
2. American Nuclear Society, October 1983, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors," ANSI/ANS-8.1-1983, La Grange Park, IL.
3. Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Waste," January 1999.
4. Code of Federal Regulations, Title 10, Part 71, "Packaging and Transportation of Radioactive Material," January 1999.
5. NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel," U.S. Nuclear Regulatory Commission, March 1998.
6. Sixth International Conference on Nuclear Criticality Safety, Versailles, France, September 20-24, 1999.
7. "Topical Report on Actinide-Only Burnup Credit for PWR Spent Fuel Packages," DOE/RW-0472, Rev. 2, U.S. Department of Energy, September 1998.
8. Spent Fuel Project Office Interim Staff Guidance - 8, Revision 0, U.S. Nuclear Regulatory Commission, May 16, 1999.
9. Chester Poslusny, "Summary of Public Workshop with the Nuclear Energy Institute on Burnup Credit," NRC Memorandum to Susan Shankman, U.S. Nuclear Regulatory Commission, May 27, 1999. (Meeting held on May 17, 1999)
10. Cecil Parks, Charles Nilsen, "NRC Research Program on Burnup Credit," 27th Water Reactor Safety Information Meeting, Bethesda, MD, October 25-27, 1999.
11. Spent Fuel Project Office Interim Staff Guidance - 8, Revision 1, U.S. Nuclear Regulatory Commission, July 30, 1999. (http://www.nrc.gov/OPA/reports/isg8r1.htm)

Table 1. Results from NRC Confirmatory Analysis of Table 7-4 in DOE Topical Report, DOE/RW-0472 Rev.2, Tally of the Use of Fission-Product and ^{236}U Margin for Addressing Uncertainties of Actinide-Only Burnup Credit

Enrichment (wt% ^{235}U) and Burnup (GWD/MTU)		EFPM = Estimated Fission Product and ^{236}U Margin (%Δk_{eff})		DOE's Reactivity Allowances for Uncertainty Issues and Approximations in Actinide-Only Burnup Credit (%Δk_{eff})			Estimated Remaining Margin (%Δk_{eff}) with EFPM from:	
		DOE TR Rev.2	NRC Case A	Criticality Validation Issues	Effect if Control Rods were Inserted During Depletion	Computer Code Adequacy Issues	DOE TR Rev.2	NRC Case A
3.0	15	8.4	4.4	2.0	3.3	1.0	2.1	**-1.9**
	30	13.0	5.9	2.0	3.3	1.0	6.7	**-0.4**
	45	16.0	6.9	2.0	3.3	1.0	9.7	0.6
3.6	15	8.2	4.3	2.0	2.1	1.0	3.1	**-0.8**
	30	12.8	5.6	2.0	2.1	1.0	7.7	0.5
	45	16.2	6.7	2.0	2.1	1.0	11.1	1.6
4.5	15	7.9	4.2	2.0	1.0	1.0	3.9	0.2
	30	12.4	5.6	2.0	1.0	1.0	8.4	1.6
	45	16.1	6.5	2.0	1.0	1.0	12.1	2.5

Appendix A

NRC Spent Fuel Project Office
Interim Staff Guidance - 8
Revision 1
(July 30, 1999)

ISG-8, Rev. 1 - Limited Burnup Credit

Spent Fuel Project Office

Interim Staff Guidance - 8

Revision 1

Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks

Introduction:

Unirradiated reactor fuel has a well-specified nuclide composition that provides a straightforward and bounding approach to the criticality safety analysis of transport and storage casks. As the fuel is irradiated in the reactor, the nuclide composition changes and, ignoring the presence of burnable poisons, this composition change will cause the reactivity of the fuel to decrease. Allowance in the criticality safety analysis for the decrease in fuel reactivity resulting from irradiation is typically termed burnup credit. Extensive investigations have been performed both within the United States and by other countries in an effort to understand and document the technical issues related to burnup credit. Much of this work has been considered in the development of the U.S. Department of Energy's Topical Report (TR) on Actinide-Only Burnup Credit for Pressurized Water Reactor (PWR) Spent Nuclear Fuel Packages (DOE/RW-0472).

The technical information provided in the literature and in the various TR revisions, together with the initial confirmatory analyses by the U.S. Nuclear Regulatory Commission (NRC) research program, have provided a sufficient basis for the staff to proceed with acceptance of a burnup credit approach in the criticality safety analysis of PWR spent fuel casks as discussed in the Recommendations below. Although insights gained from reviewing the TR submittals form a part of the basis for the staff's position, this interim staff guidance does not approve the TR or its supporting documentation. The following recommendations provide a cask-specific basis for granting burnup credit, based on actinide composition. The NRC's staff will issue additional guidance and/or recommendations as information is obtained from its research program on burnup credit and as experience is gained through future licensing activities. Except as specified in the following recommendations, the application of burnup credit does not alter the current guidance and recommendations provided by the NRC staff for criticality safety analysis of transport and storage casks.

Recommendations:

1. **Limits for the Licensing Basis.** The licensing-basis analysis performed to demonstrate criticality safety should limit the amount of burnup credit to that available from actinide compositions associated with PWR irradiation of UO_2 fuel to an assembly-average burnup value of 40 GWd/MTU or less. This licensing-basis analysis should assume an out-of-reactor cooling time of five years and should be restricted to intact assemblies that have not used burnable absorbers. The initial enrichment of the fuel assumed for the licensing-basis analysis should be no more than 4.0 wt% ^{235}U unless a loading offset is applied. The loading offset is defined as the minimum amount by which the assigned burnup loading value (see Recommendation 5) must exceed the burnup value used in the licensing safety basis analysis. The loading offset should be at least 1 GWd/MTU for every 0.1 wt% increase in initial enrichment above 4.0 wt%. In any case, the initial enrichment shall not exceed 5.0 wt%. For example, if the applicant performs a safety analysis that demonstrates an appropriate subcritical margin for 4.5 wt% fuel burned to the limit of 40 GWd/MTU, then the loading curve (see Recommendation 4) should be developed to ensure that the assigned burnup loading value is at least 45 GWd/MTU (i.e., a 5 GWd/MTU loading offset resulting from the 0.5 wt% excess enrichment over 4.0 wt%). Applicants requesting use of actinide compositions associated with fuel assemblies, burnup values, or cooling times outside these specifications, or applicants requesting a relaxation of the loading offset for initial enrichments between 4.0 and 5.0 wt%, should provide the measurement data and/or justify extrapolation techniques necessary to adequately extend the isotopic validation and quantify or bound the bias and uncertainty.

Code Validation. The applicant should ensure that the analysis methodologies used for predicting the actinide compositions and determining the neutron multiplication factor (k-effective) are properly validated. Bias and uncertainties associated with predicting the actinide compositions should be determined from benchmarks of applicable fuel assay measurements. Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that represent important features of the cask design and spent fuel contents. The particular set of nuclides used to determine the k-effective value should be limited to that established in the validation process. The bias and uncertainties should be applied in a way that ensures conservatism in the licensing safety analysis. Particular consideration should be given to bias uncertainties arising from the lack of critical experiments that are highly prototypical of spent fuel in a cask.

Licensing-Basis Model Assumptions. The applicant should ensure that the actinide compositions used in analyzing the licensing safety basis (as described in Recommendation 1) are calculated using fuel design and in-reactor operating parameters selected to provide conservative estimates of the k-effective value under cask conditions. The calculation of the k-effective value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate representation of the physics. Of particular concern should be the need to account for the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the assumed axial burnup profiles), the need to consider the more reactive actinide compositions of fuels burned with fixed absorbers or with control rods fully or partly inserted, and the need for a k-effective model that accurately accounts for local reactivity effects at the less-burned axial ends of the fuel region.

Loading Curve. The applicant should prepare one or more loading curves that plot, as a function of initial enrichment, the assigned burnup loading value above which fuel assemblies may be loaded in the cask. Loading curves should be established based on a 5-year cooling time and only fuel cooled at least five years should be loaded in a cask approved for burnup credit.

Assigned Burnup Loading Value. The applicant should describe administrative procedures that should be used by licensees to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include an assembly measurement that confirms the reactor record assembly burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For an assembly reactor burnup record to be confirmed, the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by the combined uncertainties in the records and the measurement.

Estimate of Additional Reactivity Margin. The applicant should provide design-specific analyses that estimate the additional reactivity margins available from fission product and actinide nuclides not included in the licensing safety basis (as described in Recommendation 1). The analysis methods used for determining these estimated reactivity margins should be verified using available experimental data (e.g., isotopic assay data) and computational benchmarks that demonstrate the performance of the applicant's methods in comparison with independent methods and analyses. The Organization for Economic Cooperation and Development Nuclear Energy Agency's Working Group on Burnup Credit provides a source of computational benchmarks that may be considered. The design-specific margins should be evaluated over the full range of initial enrichments and burnups on the burnup credit loading curve(s). The resulting estimated margins should then be assessed against estimates of: (a) any uncertainties not directly evaluated in the modeling or validation processes for actinide-only burnup credit (e.g., k-effective validation uncertainties caused by a lack of critical experiment benchmarks with either actinide compositions that match those in spent fuel or material geometries that represent the most reactive ends of spent fuel in casks); and (b) any potential nonconservatisms in the models for calculating the licensing-basis actinide inventories (e.g., any outlier assemblies with higher-than-modeled reactivity caused by the use of control rod insertion during burnup).

Approved

(Original Signed by)

E. William Brach

Appendix B

Presentation Handout

Spent Fuel Burnup Credit in Casks: An NRC Perspective

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27th Water Reactor Safety Information Meeting
Bethesda, Maryland
October 26, 1999

1

What is Burnup Credit?

- **Fresh Fuel Assumption:**
  - Analyzing spent fuel as though it were fresh and without burnable poisons leads to excessive requirements for neutron poisons and/or spacing of spent fuel.
- **Spent Fuel Burnup Credit:**
  - Considers the reduced reactivity of irradiated fuel as governed by the changing composition of:
    - Fissile Actinides (U-235, Pu-239, Pu-241,...)
    - Non-Fissile Actinides (U-238, Pu-240, Am-241,...)
    - Fission Products (Rh-103, Cs-133, Nd-143,...)
    - Fixed Burnable Poisons (Gd, B, Er,...)

2

International Interest in Burnup Credit

- ICNC'99 had four technical sessions devoted to Burnup Credit
- Existing or Planned Uses of Burnup Credit
  - Storage Pools - PWR and BWR
  - Transport & Storage Casks - PWR only
  - Geologic Disposal - PWR and BWR
  - Reprocessing Plants - PWR and BWR

3

U.S. Efforts on Burnup Credit in Casks (1)

- DOE Work: Topical Report on Proposed Method for Actinide-Only Burnup Credit in PWR SNF Casks
  - Topical Report went through two cycles of revisions based on NRC's review and comments.
  - NRC has not approved DOE Topical Report due to outstanding technical issues and uncertainties.

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U.S. Efforts on Burnup Credit in Casks (2)

- NRC Work: Interim Staff Guidance on Burnup Credit in PWR Spent Fuel Casks
  - March 1999: Started NRC Research Program on Burnup Credit.
  - May 1999, ISG-8 Rev.0: Approval of Limited Actinide-Only Burnup Credit based on 50% of Verified Burnup. NRC/NEI Workshop.
  - July 1999, ISG-8 Rev.1: Approval of Limited Actinide-Only Burnup Credit based on 100% of Verified Burnup. Applications expected in early 2000.
- NRC expects to issue further guidance revisions to reflect new research and licensing experience.

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Proposed Form of Burnup Credit in Casks

- Input three pieces of fuel information:
  - Fuel Assembly Design:
    - Dimensions, Initial enrichment, Internal poisons
  - Assembly-Average Burnup
  - Minimum Cooling Time
- Output a cask loading criteria curve:
  - Minimum Burnup versus Initial Enrichment

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Details of Fuel Power History Affect the Reactivity of Spent Fuel

- Parameters for PWR Fuel Power History:
  - Absorber Rods, Dissolved Boron, Moderator Temperature
  - Fuel Temperature, Specific Power
- Assemblies of given Design, Average Burnup, and Cooling Time have a wide range of:
  - Isotopic Compositions
  - Burnup Profiles - Axial, Horizontal

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Proposed Analysis Approach for Burnup Credit in Spent Fuel Casks

- For Assemblies of given Design, Average Burnup, and Cooling Time:
  - Determine burnup isotopic compositions and profiles that maximize the computed in-cask reactivity.
  - Calculate k_{eff} for a payload of such assemblies under normal and accident conditions in the cask.
- Use k_{eff} results over a range of burnups and initial enrichments to develop a Burnup Credit Loading Curve.

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Burnup Credit Modeling and Validation Color Code

- Green - Adequate for use with full burnup credit
- Yellow - Adequate for Limited Actinide-Only Burnup Credit (ISG-8 Rev.1)
- Red - Potential basis for expanded or full burnup credit

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Modeling Issues for Burnup Credit (1)

- **Fuel Isotopics: Burnup History Parameters**

Actinide-only k_{eff} is bounded by maximizing:

- Solid Poisons (Control rods, Internal poisons, etc.) [Red]
- Dissolved Boron [Yellow]
- Moderator Temperature [Yellow]
- Fuel Temperature [Yellow]
- Specific Power [Yellow]

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Modeling Issues for Burnup Credit (2)

- **In-Cask Neutronics: Horizontal Burnup Profiles within Assemblies**
  - Effects of tilted ! .nup profiles within assemblies are especially significant in small casks.
  - Most-reactive credible configuration must consider relative orientations of assemblies with strong burnup tilts.
  - DOE has proposed an acceptable method for modeling the effects of horizontal burnup profiles. [Green]

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Modeling Issues for Burnup Credit (3)

- **In-Cask Neutronics: Axial Burnup Profiles and End Effects**
  - Ends of spent fuel are less burned, more reactive.
  - Fission chain reaction is localized at least burned end.
  - Determine and assume most reactive burnup profiles.
  - DOE has evaluated axial burnup profiles based on in-core data for >3000 assemblies at several PWRs. [Yellow]
  - NRC working with NEI and OECD/NEA to further evaluate axial profiles, noding, and end-effects modeling for various neutronic conditions in casks. [Red]

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A Simple Validation Question:

- **Fact: We can calculate criticality in PWRs with known accuracy.**
- **Question: Can we somehow conclude from this that we can calculate the k_{eff} of spent fuel in casks with:**
  - (a) Similar accuracy? Answer: No.
  - (b) Some other level of accuracy? Answer: Maybe. But: 1) it's not easy; 2) benefits limited; 3) more data and analysis needed.
- **Why is this so?**
  - Neutrons see irradiated fuel in casks differently from in a reactor
  - Core calculations benefit from feedback. No feedback from spent fuel.
  - What is needed? Let's try to understand this ...

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Validation

- Method Validation = Evaluating and using the method bias and its uncertainty
- We can start to understand the "Simple Validation Question" by comparing PWR and Spent Fuel Cask Criticality in terms of:
  - Phenomena
  - Analysis
  - Methods
  - Validation

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Understanding the Validation Issue (1)

- PWR Cores - Predicting Restart Criticality
  - **Phenomena**: Whole-Core criticality in similar cores with mixed cycle burnups, designed-in power flattening, no end effects.
  - **Analysis**: Predict core criticality knowing past restart data and detailed fuel operating history.
  - **Typical Methods:**
    - Isotopics - 2D Transport, Multigroup
    - Criticality - 2D/3D Transport/Diffusion, Approximate Mesh Geometry, Multigroup/Few-Group
  - **Validation**: Accumulated PWR restart data have taught code developers and users how to predict PWR restart criticality.

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Understanding the Validation Issue (2)

- Cask Burnup Credit - Ensuring Subcriticality
  - **Phenomena:** Localized criticality at fuel ends; Variety of fuel loadings; Variety of cask designs; Unique material geometries for neutron absorption and scattering.
  - **Analysis:** Dead Reckoning - no feedback. Try to bound the maximum k_{eff}. Assume most reactive fuel power histories and burnup profiles.
  - **Typical Methods:**
    - Isotopics - 1D/2D Transport, Multigroup
    - Criticality - 3D Monte Carlo, Exact Geometry, Multigroup or Continuous Energy
  - **Validation:** (see next slides...)

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Potential Sources of Data for Validating Cask Burnup Credit Methods (1)

- **Isotopic Validation:**
  - Radiochemical assay of spent fuel
    - Actinide Data for burnups ≤40 GWD/MTU) [Yellow]
    - Actinide Data for initial enrichments >4% [Red]
    - Actinide Data for burnups >40 GWD/MTU [Red]
    - Actinide Data for fuels with internal absorbers [Red]
    - Fission Product Data [Red]
  - Nondestructive assay of spent fuel
    - Advanced/Novel Methods [Red]

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Potential Sources of Data for Validating Cask Burnup Credit Methods (2)

- **Combined Isotopic and Criticality Validation:**
  - Criticality in Power Reactors [Red]
    - Planned in DOE Topical Report for Disposal Criticality
    - Other approaches
  - Criticality in HEU Reactors [Red]
  - Subcritical Measurements on Spent Fuel [Red]

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Potential Sources of Data for Validating Cask Burnup Credit Methods (3)

- **Criticality (k_{eff}) Validation:**
  - Lab Critical Experiments
    - Existing UO_2 and MOX Data [Yellow]
    - New experiments relevant to spent fuel, end effects, etc. [Red]
  - Reactivity Worth Experiments
    - Existing and Planned Foreign or Proprietary Data (e.g., French, British) [Red]
    - Planned REBUS Experiments (Belgonucleaire) [Red]
    - Other New Experiments (e.g., Proteus, U.S. Ceres) [Red]

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (1)

- Limits for Licensing Basis:
  - Credit from Actinides Only in UO_2 PWR Fuel
  - Maximum credited burnup is 40 GWD/MTU
  - No fuel designs with burnable poisons
  - Assume cooling time of 5 years (minimum)
  - Loading offset for enrichments between 4 and 5%
  - Fuels and actinide compositions outside these limits require extension of the isotopic validation:
    - Provide measurement data, and/or
    - Justify techniques for extrapolating bias and uncertainty

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (2)

- Validation of Codes and Methods:
  - Derive isotopic bias & uncertainty from applicable fuel assay benchmarks.
  - Derive k_{eff} bias & uncertainty from benchmark experiments representing major features of cask and spent fuel.
  - In computing k_{eff}, use only those nuclides established in validation process.
  - Consider the bias uncertainties arising from lack of experiments that are prototypic of spent fuel in the cask.
  - Apply bias and uncertainties only in ways that ensure conservatism in the licensing safety analysis.

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (3)

- Model Assumptions for Licensing Basis:
  - In isotopic calculations, use the fuel design and power history parameters that maximize k_{eff} in the cask.
  - Calculate k_{eff} using models and assumptions that allow adequate representation of important physics, including:
    - The axial and horizontal burnup profiles within assemblies
    - The more reactive actinide compositions of fuels burned with inserted control rods or internal absorbers
    - Local neutron scattering and absorption effects near the least-burned end of the fuel

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (4)

- Cask Loading Curves:
  - As a function of initial enrichment, plot the Assigned Burnup Loading Value above which fuel assemblies may be loaded.
  - Loading curves based on analysis for 5-year cooling.
  - Load only assemblies cooled 5 years or more.

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (5)

- Assigned Burnup Loading Value:
  - Applicant describes administrative procedures by which cask user ensures that fuel loading is within specifications.
  - Procedures include a measurement that confirms the reactor record value of assembly burnup.
    - Measurement may be calibrated to the reactor records for a representative set of assemblies.
    - Confirmation: Measured and record burnup values agree within a 95% confidence interval based on measurement uncertainty.
  - Reduce the confirmed record value of assembly burnup by the combined uncertainties in the records and the measurement.

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ISG-8 Rev.1: NRC Guidance on Burnup Credit for PWR Spent Fuel in Casks (6)

- Estimate of Additional Reactivity Margin:
  - Estimate the additional reactivity margins from actinides and fission products not included in licensing safety basis.
  - Verify the analysis methods for estimating margins using:
    - Available experimental data (e.g., isotopic assays)
    - Computational benchmarks comparing against independent methods and analyses.
  - Assess estimated margins against:
    - Any uncertainties not directly accounted for in the modeling or validation process (e.g., non-prototypicality of k_{eff} benchmarks)
    - Any potential nonconservatisms in the licensing-basis models and assumptions (e.g., neglect of outlier rodded burnup histories)

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PWR Spent Fuel Burnup Credit in Casks: What Next?

- **Near Term Issues:**
  - Burnable Absorbers [Red]
    - Absorber designs in PWR fuels (Gd, B, Er,...)
    - Applicable isotopic validation
    - Burnup computational models -1D and 2D
  - Rodded Burnup Histories [Red]
    - At-power use of control rods in U.S. PWRs
      - **Worst-Case Plants: How, How much, Where, When?**
    - Applicable isotopic validation and modeling
  - Burnup Verification Measurements [Yellow]
  - Fission Product Margin and Uncertainties [Red]

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PWR Spent Fuel Burnup Credit in Casks: Conclusion

- NRC will issue further technical guidance on Burnup Credit as information and insights emerge from:
  - Cooperative Research
  - Licensing Experience
  - Industry Data and Analysis (as available)

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Burnup Credit For Spent Fuel Transport

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Abstract

Consideration of the "true" state of reactivity of spent nuclear fuel has been proposed for demonstrating criticality safety for casks used to transport these materials. The proposed practice, which accounts for the fuel burnup history in determining the state of reactivity for the fuel, is known as burnup credit. The motivation for this proposal is the expectation of improving packaging efficiencies of these spent fuel casks by increasing capacities, thereby, reducing the number of shipments, worker and public exposure to radiation, and costs of transport. The use of burnup credit would replace the practice of assuming spent fuel to be in its most reactive or fresh state, that is, the use of the "fresh fuel" assumption. In order to maintain criticality safety for spent fuel transport casks that use burnup credit, the ability to adequately predict the reactivity state of spent fuel must available. A method that has been proposed by DOE for performing criticality analysis for spent fuel systems is described. The method has been reviewed by the NRC, and has been used to issue general guidance on the subject. The effect of that guidance is discussed. Finally, the general implications of using burnup credit are described in terms of risk to public health and safety.

Introduction

To support development of advanced technology spent fuel transportation casks the U.S. Department of Energy (DOE) began to pursue the use of burnup credit in the mid-1980's. The approach that DOE planned to follow in its pursuit of burnup credit was first presented to the Nuclear Regulatory Commission (NRC) at a DOE sponsored workshop held in February 1988. The workshop was first of many meetings, and a great deal of correspondence between NRC and DOE on the subject of burnup credit. In 1988, the DOE strategy was to seek NRC approval of "full" burnup credit that would cover the range of initial enrichments and burnup values of all spent fuel in the anticipated inventory. Although the full burnup credit approach was not actually intended to take credit for all possible negative reactivity that could be attributed to burnup, it would take credit for the negative reactivity that would be practical. That is, it would account for all fissile actinides, most neutron absorbing actinides, and a small number of fission products that accounted for about 80% of the available credit for all fission products.

The DOE's first submittal to NRC of its proposed burnup credit methodology in May 1995 had deferred the credit for fission products [1]. The "Rev. 0" Topical Report was an actinide only approach. Although DOE ceased work on advanced technology casks by 1996, the transportation burnup credit activities continued, and NRC's review and comments on the 1995 report led to a "Rev. 1" version of the report submitted in May 1997 [2]. This version of the report provided additional support for the proposed methodology for actinide only burnup credit for spent fuel up to 5% initial enrichment and 50 GWd/t burnup. The proposal was not accepted by NRC.

Following receipt of NRC's comments on the Rev. 1 report, DOE decided to develop a final version of the actinide only approach. During development of revision 2 of the actinide only burnup credit topical report, DOE concluded that further progress in the use of burnup credit for transport of spent fuel would be more effective in applications to specific cask designs. Therefore, submittal of the revised report marked the conclusion of these activities and the start of efforts to transfer the technology to the private sector.

The final version of the DOE topical report was submitted as "Rev. 2" in October 1998 [3]. The approach proposed in Rev. 2, used limited initial enrichment and burnup of 4% and 40 GWd/t, respectively. NRC issued interim guidance on transportation burnup credit (ISG-8) on 17 May 1999 [4]. The guidance endorsed the use of a limited version of DOE's final proposal for actinide only burnup credit for transport of spent fuel. NRC issued a revision to their Interim Staff Guidance on August 8, 1999 [5].

The Method Proposed by DOE for Actinide-Only Burnup Credit

Revision 2 of the Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages describes a method for performing and applying nuclear criticality safety calculations using actinide-only burnup credit. The changes in the U-234, U-235, U-238, Pu-238, Pu-239, Pu-240, Pu-241, Pu-242, and Am-241 concentration that result from burnup are used in burnup credit criticality analyses. No credit is taken for fission product neutron absorbers.

The method described in the topical report consists of five major steps. These include validation of depletion codes, validation of criticality codes, treatment of end effects and other modeling considerations, generation of spent fuel loading curves, and loading verification procedures.

Validation of a computer code system to calculate isotopic concentrations of spent fuel created during burnup in the reactor core and subsequent decay is required. A set of 54 chemical assays are presented for this purpose of benchmarking depletion codes. The report also presents a method for assessing the calculational bias and uncertainty, and conservative procedures for applying correction factors for each isotope.

Validation of a computer code system used to predict the subcritical multiplication factor, k_{eff}, of a spent fuel cask is also required. The cask vendor is to select from available UO_2 critical experiments that would best represent the cask. Then forty-seven UO_2/PuO_2 critical experiments have been selected to validate the transuranic isotopes. The method uses the most limiting of the upper safety limits on k_{eff} (which can be a function of trending parameters) from either the UO_2 or UO_2/PuO_2 set to assure that the calculated k_{eff} when increased for the bias and uncertainty is less than 0.95.

Three bounding axial profiles are established for the isotopic concentration and criticality calculations. The three bounding profiles were established from examination of 3,169 axial profiles, to assure the "end effect" is accounted for conservatively.

The validated codes and bounding conditions are used to generate cask loading criteria (burnup credit loading curves). Burnup credit loading curves show the minimum burnup required for a given initial enrichment. The NRC licensed utility's burnup record is compared to this minimum burnup requirement after the utility accounts for the uncertainty in its record. Separate curves may be generated for each assembly design, various minimum cooling times, and burnable absorber histories.

Verification that spent fuel assemblies meet the package loading criteria and confirmation of proper assembly selection prior to loading must be performed. A measurement of the average assembly burnup is required.

Each step is described in detail for use with any computer code system and is then demonstrated with the SCALE 4.2, computer code package, using 27BURNUPLIB cross sections. However, the procedures described are intended for general use, and could be used with other calculation methods.

<u>NRC Initial Interim Guidance on Burnup Credit</u>

On 17 May 1999, at a public meeting between NRC and the Nuclear Energy Institute (NEI), the NRC presented their interim guidance on limited burnup credit (ISG-8).

- The NRC guidance allows limited partial credit for spent fuel burnup. With a few exceptions, NRC endorses the method described in DOE's, "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages," DOE/RW-0472, Rev. 2. There are, however, two notable exceptions to the method proposed in the report. The two exceptions to the method proposed by DOE are:
  - For cask loading, burnup is assumed as 50% of the verified and adjusted assembly average value.
  - Uncertainties from reactor records and verification measurements of burnup must be combined. DOE proposed only uncertainties in records be applied.

<u>NRC's Plans to Assess the Use of Burnup Credit for Spent Fuel Transport</u>

Participants at the 17 May 1999 meeting were informed by NRC's Spent Fuel Project Office (SFPO) that the guidance represented a first step in a process, intended to expand the credit for burnup. SFPO is the NRC organization responsible for licensing of spent fuel transport and storage systems. SFPO will be supported by NRC's Office of Research (RES) in their efforts to expand the credit given for burnup. The NRC described a two phased research activity to be conducted by RES to expand credit beyond initial guidance issued by SFPO in ISG-8. The research effort was described as follows:

- In the first phase, RES will evaluate available information. This effort is expected to result in a modest increase in Actinide-Only burnup credit within six months.

- A second phase effort, expected to take two to three years, will define data and experimental needs required to further expand the credit given for burnup as a result of the first phase activities. This expansion is expected to include some credit for fission products.

NRC advised that their ability to fund burnup credit research efforts was limited. Therefore, research and data acquisition needs identified to support any expansion of the burnup credit would be industry's responsibility.

The NRC guidance is in the form of a recommendation to those wishing to use burnup credit for transportation of spent fuel in an NRC approved cask. The guidance indicates that NRC has reviewed and accepted the basic methodology. The applicant must show the applicability of the method to the specific case, identify any additional information required for the case of interest, and provide the analysis done to show that the method was used correctly.

NRC Revised Interim Guidance on Burnup Credit

On August 8, 1999, the NRC issued "Interim Staff Guidance - 8, Revision 1." The revised guidance appears to allow significantly more burnup credit for PWR spent nuclear fuel than was allowed by the initial guidance. Although the revised guidance appears to give more credit for burnup, the simplicity of the original guidance has been lost. The previous version of guidance used the DOE's Revision 2 to the "Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages." Initial NRC guidance was to use 50% of what was applied for in DOE's Topical Report. In revision 1, the NRC specifically states, "Although insights gained from reviewing the Topical Report (TR) submittals form a part of the basis for the staff's position, this interim staff guidance does not approve the TR or its supporting documentation." This comment removes reliance on DOE's Topical Report, and increases uncertainty regarding what is acceptable. This section of the paper will highlight those areas of the revised guidance that seem unclear, and might benefit by further explanation or clarification in any subsequent revision of the guidance.

1. Limits for the Licensing Basis

The NRC states,

> "This licensing-basis analysis should assume an out-of-reactor cooling time of five years and should be restricted to intact assemblies that have not used burnable absorbers."

Although this is clearly stated, it is inconsistent with historical interactions with the NRC. It is believed that the NRC meant "a cooling time of *at least* five years" and "have not used *integral* burnable absorbers." These modifications would make the restrictions consistent with DOE's long-time proposals on these issues. The first, relating to cooling time, has been advocated since 1988. The second, regarding integral burnable poisons, has been declared for over five years. During discussions and correspondence with the NRC on burnup credit neither of these positions seem to have been rejected.

The NRC clearly places a limit on the credit for the burnup at 40 GWD/MTU and states that if credit for above 4 wt% enrichment is desired a 1 GWD/wt% penalty should be applied. It then allows for a reduction in this penalty if more chemical assay data are provided. It is assumed that NRC means more than that provided in DOE's Topical Report.

2. Code Validation

The NRC states:

> "Bias and uncertainties associated with predicting the actinide compositions should be determined from benchmarks of applicable fuel assay measurements."

Although not stated explicitly, it seems reasonable to interpret this as acceptance of the methods for isotopic validation given in DOE's Topical Report.

The NRC then states:

> "Bias and uncertainties associated with the calculation of k-effective should be derived from benchmark experiments that represent important features of the cask design and spent fuel contents. The particular set of nuclides used to determine the k-effective value should be limited to that established in the validation process. The bias and uncertainties should be applied in a way that ensures conservatism in the licensing safety analysis."

This again seems to be consistent with the positions taken in DOE's Topical Report which in Rev. 2 eliminated U-236 since it was not in the benchmark experiments.

Finally, the NRC states:

> "Particular consideration should be given to bias uncertainties arising from the lack of critical experiments that are highly prototypical of spent fuel in a cask."

There was no statement like this in DOE's Topical Report. DOE had suggested that this was covered by not taking credit for fission products but this may still be an issue for the NRC. This statement would suggest that the applicant follow DOE's methodology until the NRC provides more definitive guidance.

3. Licensing-Basis Model Assumptions

The NRC states:

> "The applicant should ensure that the actinide compositions used in analyzing the licensing safety basis are calculated using fuel design and in-reactor operating parameters selected to provide conservative estimates of the k-effective value under cask conditions. The calculation of the k-effective value should be performed using cask models, appropriate analysis assumptions, and code inputs that allow adequate representation of the physics. Of particular concern should be the need to account for the axial and horizontal variation of the burnup within a spent fuel assembly (e.g., the assumed axial burnup profiles), the need to consider the more reactive actinide compositions of fuels burned with fixed absorbers or with control rods fully or partly inserted, and the need for a k-effective model that accurately accounts for local reactivity effects at the less-burned axial ends of the fuel region."

This again appears to be consistent with DOE's Topical report. However, there may be items that the NRC is objecting to. For example, DOE's positions on the effect of control rods on depletion was that the control rods were generally out and the fraction of assemblies under inserted control rods is small. DOE showed that the reactivity effect on isotopic concentrations due to control rods was fairly large but covered by the fission product margin. NRC has never given approval or denial of this position. In the absence of further guidance, it would appear reasonable to assume NRC approval of DOE's position.

4. Loading Curve

The NRC states:

> "The applicant should prepare one or more loading curves that plot, as a function of initial enrichment, the assigned burnup loading value above which fuel assemblies may be loaded in the cask. Loading curves should be established based on a 5-year cooling time and only fuel cooled at least five years should be loaded in a cask approved for burnup credit."

This statement leads to some confusion, and may unnecessarily constrain loading specifications. The only reasons for more than one loading curve per fuel type were burnable absorber loading and/or cooling time. Cooling time between 5 and 15 years has a significant reactivity effect due to Pu-241 decay, and it is suggested that a minimum cooling time, not less than 5 years, be specified. Then, one or more loading curves can be established for a specific cask design for various fuel designs and cooling times.

5. Assigned Burnup Loading Value

The NRC states:

> "The applicant should describe administrative procedures that should be used by licensees to ensure that the cask will be loaded with fuel that is within the specifications of the approved contents. The administrative procedures should include an assembly measurement that confirms the reactor record assembly burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. For an assembly reactor burnup record to be confirmed, the measurement should provide agreement within a 95 percent confidence interval based on the measurement uncertainty. The assembly burnup value to be used for loading acceptance (termed the assigned burnup loading value) should be the confirmed reactor record value as adjusted by reducing the record value by the combined uncertainties in the records and the measurement."

This is consistent with DOE's Topical Report with the exception of using the combined uncertainties of the records and the measurement. It is recommended to statistically combine these uncertainties. This would essentially make the uncertainty the same as that of the measurement device since it is expected to be larger than the uncertainty of the reactor records. The NRC was helpful in clarifying that the basis for burnup is the reactor record burnup and that the calibration of the measurement device can utilize the reactor records. Both of these issues had been left unresolved in DOE's Topical Report.

The wording in this guidance instruction seems to suggest that the cask designer or vendor will perform burnup verification measurements. This was specifically avoided in DOE's approach because loading will be done at a reactor facility by the utility under its Part 50 license. It is recommended that this instruction be modified to reflect the separate roles of the cask designer/vendor and the reactor operator/licensee. The cask designer who applies for a certificate of compliance should provide the general requirements of performing such measurements, but the detailed test specifications should be the responsibility of the reactor operator/licensee.

6. Estimate of Additional Reactivity Margin

The NRC states:

> "The applicant should provide design-specific analyses that estimate the additional reactivity margins available from fission product and actinide nuclides not included in the licensing safety basis. The analysis methods used for determining these estimated reactivity margins should be verified using available experimental data (e.g., isotopic assay data) and computational benchmarks that demonstrate the performance of the applicant's methods in comparison with independent methods and analyses. The Organization for Economic Cooperation and Development Nuclear Energy Agency's Working Group on Burnup Credit provides a source of computational benchmarks that may be considered. The design-specific margins should be evaluated over the full range of initial enrichments and burnups on the burnup credit loading curve(s). The resulting estimated margins should then be assessed against estimates of: (a) any uncertainties not directly evaluated in the modeling or validation processes for actinide-only burnup credit (e.g., k-effective validation uncertainties caused by a lack of critical experiment benchmarks with either actinide compositions that match those in spent fuel or material geometries that represent the most reactive ends of spent fuel in casks); and (b) any potential nonconservatisms in the models for calculating the licensing-basis actinide inventories (e.g., any outlier assemblies with higher-than-modeled reactivity caused by the use of control rod insertion during burnup)."

This is a requirement that was not in DOE's Topical Report. DOE had intended that its analysis showing margin would suffice for all future applicants. Since the list of activities is similar to that performed by DOE in the Topical Report, it seems as though it would be reasonable to follow DOE's approach supplemented by performing the OECD working group benchmarks.

7. Summary

Although the NRC did not accept DOE's Topical Report on burnup credit, the guidance that the NRC gave is sufficiently similar to DOE's Topical Report that one can estimate that the burnup credit benefits are about the same.

<u>An Industry Perspective on the Risk of Using Burnup Credit</u>

The NRC's issuance of guidance on the use of burnup credit, and their continued work to expand the credit allowed, is welcomed. As we move away from using only the overly conservative fresh fuel assumption to a burnup credit option, it is appropriate to examine the effects of this change on the overall risk of transporting spent nuclear fuel. The following section of the paper has been prepared by the Electric Power Research Institute (EPRI), an organization that represents the research and development interests of the electric power industry in the U.S. The EPRI analysis examines the risks associated with using burnup credit. The quantitative estimate of risk associated with using burnup credit is shown to be negligibly small. Although not estimated quantitatively, it is noted that the risk reduction of using burnup credit that is realized by the expectation of fewer shipments when burnup credit is used, is sizable.

The NRC Transportation regulations require consideration of the effect of fresh-water in-leakage for criticality analysis of a single package for transport of fissile material [6]. The regulations do not preclude the use of burnup credit in demonstrating subcriticality, they simply require that the package (i.e., cask and contents) is shown to be subcritical.

Current regulatory practice, which has been applied to all currently approved spent fuel transport casks, does not account for the reduced reactivity of spent fuel in demonstrating subcriticality under prescribed regulatory conditions. The rules and practices currently applied to NRC Certified transport casks are listed below:

(1) Subcriticality is assured, that is, $k_{eff} < 1$.
(2) Moderation by water occurs to the most reactive credible extent.
(3) Full reflection of the system on all side by water occurs.
(4) The system is in its most reactive credible configuration consistent with the chemical and physical form of the material.
(5) The allowed k_{eff} is then reduced from 1 to account for such things as modeling and calculational biases and uncertainties
(6) Additionally, the allowable k_{eff} is further reduced by applying an arbitrary criticality safety margin of 5%, i.e., $k_{eff} = -0.05$
(7) The fuel is assumed to be in its most reactive state, which is generally unburned. This is known as the fresh fuel assumption.

Items (1) to (4) above are regulations. Item (5) is simply an acceptable approach to criticality safety analysis. Item (6) is an arbitrary safety factor applied to analysis. Item (7), the fresh fuel assumption, is the practice that is modified when burnup credit is used. It should be noted that the word "modified" not "eliminated is used. The modified item (7) might read, "The fuel is assumed to be in its most reactive arrangement, after credit for the spent fuel's burnup is determined using conservative depletion analysis."

Burnup credit only seeks a change in the fresh fuel assumption.

The issue of "Burnup Credit" (BUC) vs. the "Fresh Fuel" assumption for the evaluation of PWR spent fuel reactivity in a transportation package involves a trade-off. On one hand, consideration and credit for the reduced reactivity of the spent fuel allows for a better utilization of the package volume; this results in a greater number of assemblies per package, and, in turn, in a smaller number of shipments. On the other hand, the fresh fuel assumption provides additional margin for criticality considerations as it leads to the addition of engineered poisons within the package cavity.

Can any incremental reduction in criticality likelihood (and subsequently risks) be justified against the reduction in transportation risk deriving from using burnup credit?

NRC-sponsored work [NUREG/CR-4829, referred to as the "Modal Study") discusses the likelihood of a rail cask accident with a greater than 2% strain coupled with a concurrent submersion (Subsection 9.3.2.4). Rail shipping is particularly relevant because it is required for the dual-purpose systems that are or will be implemented at reactor sites. Under the rail shipping scenario of the Modal Study, "… this type of accident is estimated to occur once every ten million years."

The estimated frequency of a criticality event is then obtained by multiplying the frequency of the accident referred to above [i.e., 10^{-7}/year] by the likelihood that the specific package involved in the accident contain enough reactivity under the moderation and geometric conditions of the accident to result in a critical configuration [i.e., 10^{-x}/accident].

Assuming that the package system is a BUC-designed system, such a likelihood, i.e., 10^{-x}/accident, would be acceptably low if:

- Non-conservative errors associated with the specific BUC methodology are smaller than the sum of (i) the administrative margin (k_{eff} = 0.05) and (ii) the systematic bias in k_{eff} introduced in the methodology to account for enveloping conditions and uncertainties.

- The potential for human errors is small enough to protect against non-conservative fuel assembly insertion errors (misloadings)

Based on probability data for human error [Homes & Narver in NSS-8191.1, *Transportation Accidents Risks in the Nuclear Industry*], and given that (i) misloadings can introduce less reactivity as well as more reactivity, (ii) only misloadings in specific cask or canister locations have a marked effect; and (iii) two checks are required for every fuel movement, it can be estimated that the probability of a non-conservative misloading can be as high as 10^{-3} and as small as 10^{-5} for a large package. In addition, past analyses have shown that more than one misloading is required to approach criticality conditions. This brings the likelihood of having to deal with a critical configuration, given a severe enough accident, to an estimated $(10^{-3}$ to $10^{-5})^n$/accident, where "n" is the required number of non-conservative misloadings. Using the conservative assumption that only two non-conservative misloadings are required, the likelihood is (10^{-6} to 10^{-10})/accident.

Given that the frequency of an accident severe enough to result in significant damage to the package (coupled with submersion) is already very low [10^{-7}/year], the expected frequency of a critical configuration under the rail shipping scenario of the "Modal Study" is essentially zero (10^{-13} to 10^{-17}/year is a meaningless number!) Estimates of the consequences of a criticality accident are inconsequential from a risk standpoint.

Therefore, the fresh fuel assumption results in a negligible numerical reduction in critical configuration likelihood. On the other hand, by using a BUC approach, the reduction in the number of shipments is real, and results in a measurable reduction in risk associated with incident free exposures to workers and public, fatalities and injuries from radiological and non-radiological consequences of accidents, and property damage.

A risk-informed approach would seek an overall reduction of the risks associated with spent fuel shipments. A conservative approach would be to adopt the DOE methodology documented in DOE/RW-0472. This methodology would, in a cost-effective manner, deliver a substantial fraction of the benefits to be derived from using burnup credit. This methodology still does not include credit for the poisoning effect of fission products, and, therefore, a very significant margin against the potential of a criticality configuration is built into it. The technical basis for the DOE Transportation Burnup Credit methodology will be presented.

Conclusions

The recent guidance on the use of burnup credit issued by the NRC, and its promise to continue to study and expand the allowance of burnup credit for spent fuel transport is welcomed. The most recent guidance, although increasing the credit allowed over the initial guidance, is more complicated, and in some cases ambiguous. This could lead to misunderstanding by potential users of the guidance. The comments and suggestions offered in this paper, by individuals and organizations who have been involved in the process of developing a burnup credit methodology to meet regulatory requirements and the needs of users, is offered as constructive criticism. Although the NRC has closely followed the development of the methods proposed for implementing burnup credit, they have not had the advantage of the users perspective. The authors hope that the suggestions and recommendations offered here will be used by the NRC as they continue to expand and refine their guidance.

In a few years the Department of Energy will begin a thirty year program of shipping approximately 63,000 metric tons of spent nuclear fuel from commercial nuclear reactors to a first deep geological repository. The use of burnup credit offers an opportunity to reduce the numbers of shipments required to move this material by 30% to 50%. The benefits of reducing risks associated with transport is obvious. The analysis provided by EPRI in this paper suggests that the increased risk of a criticality associated with properly using burnup credit is negligible. Comparing this negligible risk component with the reduction in overall risk associated with transport supports using burnup credit.

REFERENCES

(1) Department of Energy, *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages*, May 1995 (DOE/RW-0472 Rev. 0).
(2) Department of Energy, *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages*, May 1997 (DOE/RW-0472 Rev. 1).
(3) Department of Energy, *Topical Report on Actinide-Only Burnup Credit for PWR Spent Nuclear Fuel Packages*, September 1998 (DOE/RW-0472 Rev. 2).
(4) Nuclear Regulatory Commission, Spent Fuel Project Office, Interim Staff Guidance - 8, Limited Burnup Credit, ISG-8, May 17, 1999.
(5) Nuclear Regulatory Commission, Spent Fuel Project Office, Interim Staff Guidance , ISG - 8, Revision 1, Issue: Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks," August 8, 1999.
(6) Nuclear Regulatory Commission, Title 10 Code of Federal Regulations (CFR) Part 71.

TAKING BURNUP CREDIT INTO ACCOUNT IN CRITICALITY STUDIES : THE SITUATION AS IT IS NOW AND THE PROSPECT FOR THE FUTURE

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Abstract

As the enrichment of the fuel has become higher than the limits used at the designing stages, it seemed necessary to consider the fuel depletion during irradiation to guaranty the criticality safety for highly enriched fuels transportation, storage or reprocessing. For that purpose, a method was developed considering partial Uranium-and-Plutonium burnup credit in the criticality studies ; this method was accepted by the French Safety Authority. Moreover, in order to reduce again the reactivity of irradiated fuels, a French working group was set up in 1997 to define a conservative method which enables industrial companies to take burnup credit into account with some of the fission products and using a more realistic axial profile. This work is supported by experimental programs related to the validation of the fission products effects, in terms of reactivity.

1. Introduction

Most of the facilities using irradiated fuel were originally designed for uranium oxide enriched up to 3.5% in ^{235}U. As nuclear reactor management has improved, the initial enrichment of the oxide has increased and exceeded this limit. However, the improvements that have been achieved with depletion codes and the knowledge of actinide properties have made it possible to consider the decrease in reactivity due to the burnup of the fuel.

Thus, facilities were designed taking into account a certain amount of enrichment. If more enriched fuel is used, a minimum burnup is required for the reactivity level to drop to the value used at the designing stage. In the early 80's, a method was devised, using actinides in the calculations, to define this minimum burnup. In 1997, the main companies involved in the nuclear fuel industry (Cogema, EDF, Framatome, Transnucléaire and CEA) and the Institut de Protection et de Sûreté Nucléaire (IPSN) set up a working group to determine a calculation method which makes it possible to take actinides plus some of the fission products into account in the criticality studies.

The present paper is divided into three parts. The first one presents the current French practice. The second one is related to the French Working Group : the calculations of the spent fuel composition, the characterisation of an upper-bound history of irradiation and the definition of the model used for the calculation in the criticality studies. The third one describes the various French Programs concerning the experimental validation of the criticality calculations using fission products.

2 Current French practice

Currently, criticality studies, which include irradiated fuel, consider only the heavy nucleus composition for a given specific burnup. The main isotopes of uranium and plutonium taken into account are : ^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu and ^{242}Pu. Fission products are disregarded. The calculation scheme used was validated through an experimental program (known as HTC program) carried out at the IPSN's criticality laboratory in Valduc, in collaboration with Cogema, between 1988 and 1991. The experimental configuration programs used special MOX-based fuel rods, with no fission product, and heavy nuclei compositions equal to those found in PWR fuel rods at an initial enrichment of 4.5% in ^{235}U and irradiated at 37.5 GWd/t.

The first paragraph is devoted to the history of burnup use in criticality studies. The following paragraph presents the method used when considering burnup in the studies. The third one is related to the "HTC program" set up to validate the calculation tools. The latest one shows how burnup has been taken into account in a specific example. The method has been validated by the French Safety Authority.

2.1 History

As far back as 1981, at Cogema's UP2 400 plant in La Hague, the minimum specific burnup requirement was taken into account to demonstrate the safety-criticality of reprocessing operations in the HAO facility of the UP2 plant. This plant used uranium oxide fuel, which were initially enriched to 4.2 % in ^{235}U and burnt in the CNA (B) reactor. All of these operations took place under conditions usually reserved for fuel enriched to 3.5%.

This practice was subsequently extended to other types of highly enriched uranium oxide-based PWR fuels to demonstrate the safety of their transport, interim storage and reprocessing.

2.2. *Assumptions concerning the Burnup axial profile*

This section describes the calculation method used to estimate the effective multiplication factor of interim pool storage of spent fuel assemblies and presents successively all of the assumptions made concerning geometrical modelling and the axial values of the burnup.

Figure 1 is a horizontal cross section of the 3D representation of an interim storage of spent UOX fuel assemblies. The model was fed into the MORET III(1) Monte Carlo code to calculate the effective multiplication factor of that storage configuration. The planar array, the height of the fuel assemblies and the cells were taken infinite.

As neutrons leak in a reactor around the edge of the core, leading to non-uniformity of the flux, axial and radial variations in the specific burnup of the fuel assemblies appear. However, the model used in the criticality calculations is a flat irradiation profile, with a specific method to link the burnup used in the studies and the burnup of the real fuel : the value of **the average specific burnup obtained over the 50-least-irradiated centimeters** of the fuel assembly fissile column has to be higher than the burnup used in the flat-profile model. The conservative nature of this method was discussed in a paper published at ICNC' 87 in Tokyo, Japan.

Figure 2 shows the modelling assumption used in the criticality calculations for the specific burnup shape. Height H is the height of the fuel assembly fissile column.

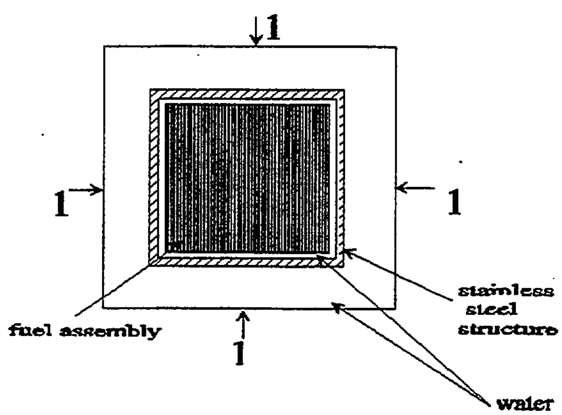


figure 1

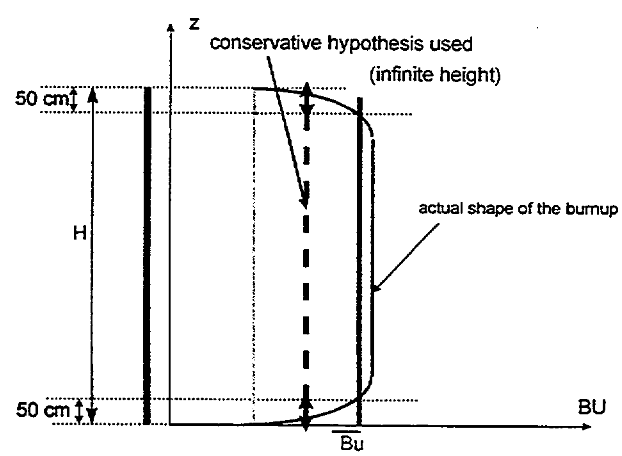


figure 2

For a given specific burnup, the composition of the fuel is computed using an irradiation history, which leads to conservative figures of uranium and plutonium isotopes concentrations. The cooling time is considered as nil.

2.3. *The validation of the calculation tools*

In order to qualify the calculations on spent fuel, IPSN carried out in 1988-1991 the HTC program, in Valduc facility. HTC "Haut Taux de Combustion" is the French name for high burnup fuels. This program concerns only Uranium and Plutonium worth in spent fuel : no fission products were taken into account. The composition of the rods especially made for the experiments (about 2500 rods) corresponded to a spent fuel initially enriched at 4.5% of ^{235}U and with a burnup of 37.5 GWd/t, but with no fission products. The ratios are given above :

$U_t/(U_t+Pu_t) = 98.9\%$	$^{239}Pu/Pu_t = 59.227\%$
$^{235}U/U_t = 1.57\%$	$^{240}Pu/Pu_t = 14.337\%$

This experimental program was performed in the "apparatus B". This apparatus consists of a water-pool with a basket. For this experiment, the basket contained the core filled with HTC rods. The water (moderator and reflector) is introduced from the bottom of the pool and the critical parameter (the height of the water in the fissile array) is determined by means of extrapolation during the sub-critical approachs.

Four types of cores were studied to investigate the spent fuel behaviour in different media, which are : a basic case (array in pure water), a dissolution medium, a pool storage medium and a shipping cask. Thus, the following different configurations were analysed :

- a single array in pure water ; five different square pitches were considered to study different moderation ratios,
- a single array in water plus boron (or gadolinium) at different pitches,
- four assemblies in absorbing canisters or not, in pure water,
- four assemblies in absorbing canisters, the whole surrounded by steal or lead screens.

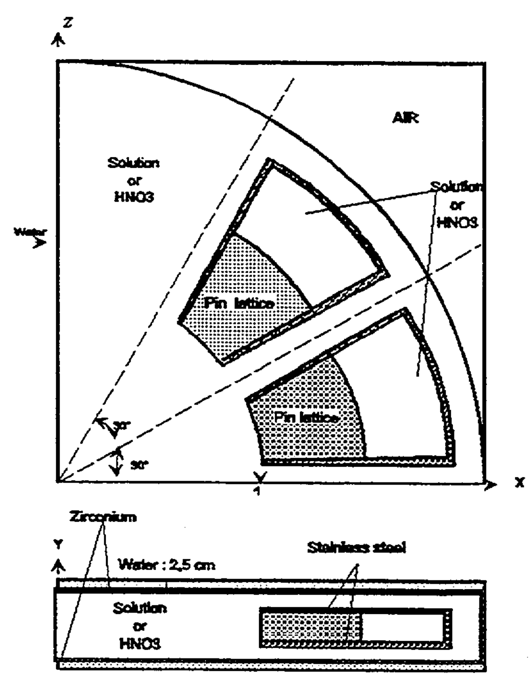
Finally, the comparison between the calculated values and the experimental ones showed a good agreement. The results are currently under revaluation with the new calculation tools.

2.4 *Example of the Continuous dissolver at La Hague*

A minimum specific burnup for fuel assemblies reprocessed in Cogema's UP3 and UP2-800 plants was taken into account at the dissolver design stage. The device design is a compromise between the capacity of the dissolver and the criticality requirements.

The dissolver comprises a flat tank fitted with a rotating section of 12 buckets. The buckets are loaded with fuel one-by-one and then, due to successive rotation of the moving part, are dived into a nitric acid solution, which is continuously renewed in the tank.

Cogema's charts gives the maximum weight of fuel allowed in each bucket for a given initial ^{235}U enrichment and a given specific burnup. These weights are then converted into lengths for each type of fuel assembly. The length therefore becomes the true parameter to control weight.



Thus, the maximum weight authorised for each bucket of fuel intended for reprocessing is determined on the basis of its initial ^{235}U enrichment and of the measured average value of the 50-least-irradiated centimeters, which is the reference value as for the interim storage case.

To illustrate the burnup effect on the management of a reprocessing plant, the weight of non-irradiated PWR UO_2 fuel enriched at 3.5% ^{235}U should be limited to 60 kg of oxide per bucket, whereas the oxide mass is not limited since the average burnup of the fuel is higher than 16 000 MWd/t over the 50 cm at both ends ; this is an usual burnup value for irradiated fuels. In these conditions, the capacity of the dissolver is not limited by criticality safety requirements.

Thus, before shearing, the specific burnup is measured along the entire length of each fuel assembly and the result is recorded. It constitutes a unique database of information on the irradiation profiles of spent fuel assemblies in boiling water or pressurised water reactors.

3. Program for the new working group

The main priority of the French working group is to study PWRs' UO_2 spent fuel assemblies.

As it has already been mentioned, only the main uranium and plutonium isotopes are taken into account in the criticality studies. The depletion code predicts the isotopic content of spent fuel (actinides and fission products), by simulating the absorption of neutrons, their fission, and the isotopes' decay in a fuel cell during irradiation in a reactor.

Since the loss of reactivity due to the fission products absorption is not taken into account, the assumption of actinide-only burnup credit leads to considerable safety margins. Then, a less conservative assumption on the fuel composition, by considering some of the fission products, would reduce the costs of transportation and storage. But, the upper-bound nature of such a method still has to be demonstrated.

Moreover, the assumption of a constant value of the burnup on the whole length of the fuel, equal to the mean value of the burnup on the 50-least-irradiated-centimeters of the fuel assembly, seems very pessimistic for most of the fuel assemblies.

Then, the French working group aims to determine a conservative method for dissolution, transportation and interim storage of spent fuel when fission products are considered and when a more realistic (so a less pessimistic) axial profile is taken into account. Four sub-groups were formed to carry out the tasks.

The first group is related to the depletion computer code results and analyses the discrepancies between calculations and measurements of the fuel composition ; it will then determine correction factors for the different element concentrations.

The second one classifies the burnup axial profiles into groups and selects some upper-bound profiles.

The third one determines the upper-bound irradiation histories for spent fuel assemblies.

The last one has to define a calculation scheme for transportation and storage criticality studies when taking burnup into account; it is especially aimed to determine the number of axial zones needed for the calculations.

3.1 *First Task : Calculation of spent fuel assembly composition*

The depletion code used for that purpose is the CESAR (2) computer code, Version 4. This version gives the concentrations of 42 heavy nuclei, 204 fission products and 100 activation products. CESAR has the advantage of being supported by a large set of qualification experiments, based on the results of representative analyses of average concentrations in fuel assemblies and on spent fuel assembly samples.

CESAR is a simplified code, which extrapolates values that have already been tabulated. The tabulated values are now prepared with APOLLO 1(5) depletion calculations.

For PWRs the experimental values of the actinides and fission products concentrations come from (i) punctual destructive analysis of PWR EDF 17x17 samples realised in a joint-venture between FRAMATOME, EDF and Cogema, (ii) analysis of the mean concentration of the dissolutions realised at La Hague - Cogema (it concerns PWR 14x14 up to 17x17 assemblies). For those samples, the burnup goes up to 60 GWd/t.

The discrepancies between calculated values and experimental ones have been analysed to determine whether they are due to cross-sections libraries, nuclear data, calculation models or experimental uncertainties. For Uranium and Plutonium isotopes, the differences between the calculated values and the experimental ones are less than 5% ; the large discrepancies met for the isotopes of Americium led to a re-evaluation of the experimental data. For fission products, differences between calculated and experimental values are less than 10%.

Gathering those results, correction factors for the concentration of every isotope will be established. For that purpose, prediction interval technique will be used : this technique defines an interval, around the mean prediction, in which there is a certain level of confidence that the next value observed will be within that interval.

3.2. *Second Task : Determining conservative specific burnup profiles*

The aim of this sub-group is to determine upper-bound specific burnup profiles for criticality calculations related to transport, interim storage and reprocessing of spent fuel assemblies.

By definition, an upper bound profile would lead to the most reactive situation, for a given average burnup and a given configuration.

The work carried out by this sub-group is primarily based on experience feedback from operators of power reactors and fuel recycling plants. This work is based on direct or indirect physical measurements of burnup made by companies involved in the fuel cycle.

The different types of information taken into account in the studies are (i) the specific burnup profiles measured in the fissile material recycling plants of the Cogema facility at La Hague, (ii) the irradiation profiles determined on the basis of EDF periodic in-service measurements of the axial and radial neutron flux in the reactors, (iii) the results and conclusions of national and international studies which have already been carried out in this area.

Measurements made at La Hague Cogema

Before dissolving the fuel assemblies in the head-end units of recycling plants UP2-800 and UP3 at Cogema's La Hague plant, the average specific burnup is measured over the entire length of the fissile column and over the 50-least-irradiated centimeters. Gamma ray spectrometry and neutron spectrometry are used to measure the average burnup of the fuel assembly. The irradiation profile is known through gamma ray spectrometry.

Cogema has already measured about 7000 irradiation profiles of fuel assemblies from PWRs with 18x18, 17x17, 16x16, 15x15 and 14x14 arrays and from BWRs with 6x6, 7x7, 8x8 and 9x9 arrays. To begin with, only the irradiation profiles of PWR fuel assemblies are being analysed.

The results will be used in two different ways : first of all, they will point out the parameters which have a considerable effect on the burnup profile, then they will give an upper bound burnup profile for every group identified. The work is in progress.

Determining profiles with theoretical studies

EDF knowledge in power reactor operations enabled them to develop validated calculation tools to accurately predict the fuel changes during irradiation. In addition to these forecasting tools, the neutronic and thermodynamic characteristics of the core are monitored by periodically inserting measurement probes into the pressure vessel and into the very heart of the reactor fuel assemblies.

All those data make it possible to determine different groups of profiles using EDF expertise and its calculation tools. The results presented below have been achieved at EDF and are described more precisely in the paper called **"Search for an Envelope Axial Burnup Profile for Use in PWR Criticality Studies in Burnup Credit"** for ICNC'99.

The study is divided in 4 main parts (i) the effects of the irradiation condition on the isotopic composition, (ii) the background related to irradiation histories of assemblies in reactors, (iii) the definition of different axial profiles for penalising irradiation histories and, finally, (iv) the determination of a conservative profile for a wet storage.

Since the burnup is mainly due to the flux level, the control rods position has been considered as the main parameter to determine the different burnup profiles. The presence of the control rod modifies the flux level at different height of the assembly ; it also hardens the flux in the assembly.

Analysing the fuel assemblies records in the reactors, EDF showed that the flux maps finally couldn't give interesting results as the least irradiated parts of the assembly are more reactive than an average-irradiated one and, then, will have a flux which is not representative of the irradiation history. Moreover, the feedback information for more than 1700 assemblies histories records pointed out that some assemblies have had 2 cycles (at least one assembly underwent 3 cycles) with the control rods inserted.

Thus, in order to define the different types of profiles, the calculations have been carried out considering the control rods inserted at a given position during 4 cycles. With the compositions achieved, calculations have been made with TRIPOLI 4 code for a single assembly surrounded by 20 cm of water.

The most conservative profile, in terms of reactivity, is not the most distorted one : there is a competition between the gain in reactivity due to a slight irradiation at the top of the assembly and the neutron leakage.

Comparison between measured and calculated profiles

The main advantage of the method is that it defines groups of specific burnup profiles using different approaches. This method will reduce the importance of the bias due to the assumptions made in each approach.

The upper-bound profiles defined above will be compared to those obtained in other studies. Moreover, for every group of burnup profiles, the margin, in terms of reactivity, will be defined when using the upper-bound profile instead of an other. The upper-bound profiles will also be compared from one group to another.

The aim of that subgroup is to provide to all of the other subgroups upper-bound specific burnup groups, with their validity ranges and the recommendations on how they should be used.

3.3. Third task : Upper bound irradiation history for criticality

The aim of this sub-group is to list and prioritise, in terms of degree of conservatism, the reactor operating stages, which should lead, for a given burnup, to differences in isotope concentration used in the criticality studies. Then, an irradiated history will be defined, which maximises the reactivity of the assembly for a given burnup.

The parameters that have been analysed are : specific power, boron concentration, down-periods and moderator temperature. The computer code used is the APOLLO 2 code with CEA 93 library.

Effects of the conditions of irradiation

For uranium oxide fuels enriched at 3.7%, Enriched Reprocessed Uranium (URE) and for gadolinium fuels, the studies have shown that, after 4 cycles of irradiation (i) the reactivity of the spent fuel increased with the specific power, the boron concentration and the moderator temperature, (ii) the reactivity decreases when the shutdown periods increase.

Those tendencies are the same after 1 cycle of irradiation, except for URE and for gadolinium fuels, which present a slight increase in the reactivity when the moderator temperature decreases : the neutrons are better-moderated and therefore lead to a more important consumption of the neutronic poisons (^{236}U for URE and Gd for Gadolinium fuels). This effect disappears after one cycle of irradiation.

However, the effects on the reactivity of the assembly are always lower than 0.8%.

Origin of the effects

Since concentrations are controlled by time-dependant phenomena (radioactive decay) and production/disappearance reactions under flux, a list will be made of the various situations involving time or influencing flux (level, energy form etc.) with their effects on concentrations.

For a given burnup, an increase of the irradiation period, which gives an increase in the flux level, leads for actinides to (i) an increase of the ^{241}Pu depletion to ^{241}Am (absorber), (ii) a decrease of the ^{239}Pu due to the ^{239}U depletion (this is an end-of-life effect). For fission products, it gives (i) an increase of ^{155}Gd as more ^{155}Eu is depleted, (ii) an increase of ^{147}Sm which compensates the decrease of ^{149}Sm (on one hand if the flux decreases, the number of absorption of ^{147}Pm decreases and so the ^{147}Sm increases ; on the other hand, as the number of absorptions of ^{147}Pm (giving ^{148}Pm) decreases, the number of ^{149}Pm decreases and, then, the number of ^{149}Sm decreases) .

For a given burnup, if the spectrum is more thermalized then (i) for actinides, the fissile isotopes will decrease (there will be fewer absorption of ^{238}U and more absorptions of ^{235}U) and the plutonium isotope concentrations will decrease too (the decrease of the absorbers will be less important than the decrease of the fissile isotopes), (ii) for fission products there will be a decrease of their concentrations with neutrons absorption during the irradiation (except for fuel with burnable poison, this effect is negligible as it is less important than the decrease of the fissile isotopes).

Conclusion

The conservative assumptions for the history of irradiation lead to the <u>hardest spectrum</u> and the <u>shortest time of irradiation</u> : the boron concentration will be equal to the maximum boron concentration, the moderator temperature will be the outlet temperature, the specific power will be equal to the maximum nominal specific power, a single-cycle of irradiation will be considered.

If no cooling time is considered after reactor shutdown, the ^{239}U will conservatively be converted into ^{239}Pu.

For a single assembly surrounded by 20-cm of water, the increase of reactivity due to those assumptions is about 1500 pcm. Furthermore, the margin between a real irradiation history and those conservative assumptions plus the conservative profile defined by EDF (as described above), will now be determined at FRAMATOME with the SCIENCE 3D-calculation scheme.

3.4 Fourth task : Modelling the specific burnup profile

The aim of this sub-group is to use the upper bound profiles achieved by the first sub-group to determine a modelling method suitable when considering burnup profiles in criticality studies for transport and interim storage of spent fuel assemblies. The assumption of a flat average distribution equal to the average value of burnup over the end 50-cm will be abandoned in favour of a multi-segment model. The work will therefore focus on determining the number **N of zones** and each zone **length**.

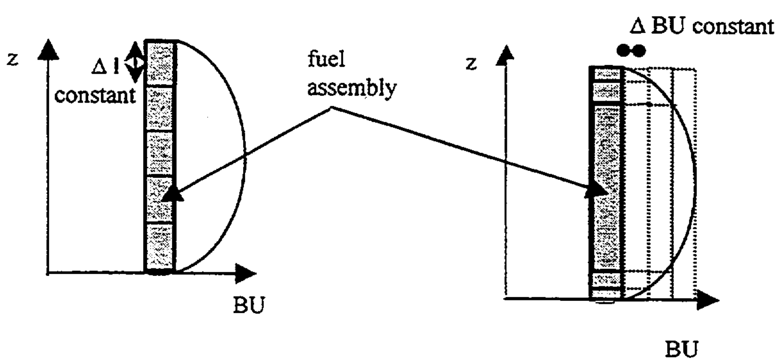
For a given configuration, the k_{eff} will be plotted versus the number of zones. The number N is set when an increase in the number of zones does not imply any significant variation of the k_{eff} value.

Calculation assumptions

The study will focus on three actual burnup profiles, concerning 17x17 PWR arrays. Initial ^{235}U enrichment is 4.5%. The first profile used corresponds to fuel assemblies which have undergone four-cycles in the reactor (average irradiation standing at about 40 GWd/t) ; the two others are related to fuel assemblies which have undergone only two-cycles in the reactor (average irradiation standing at about 20 GWd/t) and one of them was irradiated with control rods inserted.

The actinides used in the calculations are **^{235}U, ^{236}U, ^{238}U, ^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu and ^{241}Am.** The fission products used for the calculations are : **^{103}Rh, ^{133}Cs, ^{143}Nd, ^{149}Sm, ^{152}Sm and ^{155}Gd.** Together these six fission products give 50% of the reactivity loss produced by all fission products. Nine other fission products could be added to the previous list : ^{95}Mo, ^{99}Tc, ^{101}Ru, ^{109}Ag, ^{147}Sm, ^{150}Sm, ^{151}Sm, ^{145}Nd and ^{153}Eu. This group of 15 fission products accounts for around 80% of the reactivity loss produced by all fission products.

Two discretizations of the burnup profiles have been employed (they are described in the figures below) : using equal intervals of Δl or using equal intervals of $\Delta(BU)$. The computer codes used are CESAR 4, APOLLO 2[3], MORET 4 (the two last computer codes are part of the CRISTAL[4] package).



Results

Up to now, four different configurations have been studied, they concern : (i) a single assembly surrounded by 20 cm of water, (ii) an interim storage of 9-cavities-baskets with an assembly which accidentally fell nearby, (iii) an interim storage of 9-cavities-baskets in which assemblies have been moved off center and, finally, (iv) an interim storage of 9-cavities-baskets in which assemblies stand higher than the borated internal structures.

The calculations showed that, for the axial profiles used, a 13-zone discretization seemed to be sufficient.

For a burnup of 40 GWd/t, the gain on k_{eff} (k_{eff} decreases) due to fission products is between 6% and 7% with 15 fission products and between 4% and 5% with 6 fission products.

The profiles, defined by EDF within the frame of the second sub-group, will now be studied for the configurations described above and for some transportation cases. Some special libraries for CESAR 4 will be created to take the presence of control rods into account.

4. The "Fission Products" programs

The French working group gives the method that could be used to take burnup into account. Nevertheless it is necessary to know how codes can correctly predict the reactivity when fission products are considered.

For that purpose, two experimental programs are currently being run at IPSN and at CEA in collaboration with Cogema.

In 1991, the IPSN designed and carried out a preliminary series of experiments involving one of the fission products, **^{149}Sm,** in order to qualify the criticality calculation system. So that this qualification can be extended to the other five fission products selected (**^{103}Rh, ^{133}Cs, ^{143}Nd, ^{152}Sm and ^{155}Gd**), the IPSN and Cogema have joined forces to draw up and finance experiments in the context of a Common Interest Program, based on a gradual qualification strategy. The aim is to qualify the CRISTAL new criticality calculation system, by means of experiments involving the 6 fission products. The experiments will be carried out at the IPSN's criticality laboratory at VALDUC. The principle of the experiments is a sub-critical approach to find the water critical level. The advantage of this type of experiment is that the experimental set-up is similar to the configurations usually encountered in the transport, interim pool storage and dissolving of spent fuel assemblies.

An other program intends to measure the effect, in terms of reactivity, of the most important nuclei for "Burnup Credit". It involves oscillation experiments carried out in the Minerve reactor (CEA) at Cadarache, which can be used to validate the effective cross-sections and the loss of reactivity due to each fission product in the "Burnup Credit". The principle of the experiments is to measure the reactivity changes caused by slight disruptions in the neutron equilibrium in the Minerve reactor when an experimental fuel rod containing a sample (for example, a UO_2 pellet containing a selected fission product) is inserted into the centre of a fuel rod array. The results are now available and the calculations are in good agreement with the measured values. They are presented in the paper titled **"Burnup Credit for Fission Product Nuclides in PWR (UO2) Spent Fuels"** for ICNC'99. Those experiments should now be analysed to determine whether it is necessary or not to use some additional correction factors on the fission products concentrations or to define a calculation margin that should be kept.

An additional program aims to qualify the abundance of nuclei responsible for most of the "Burnup Credit". This program is based on chemical analysis of spent fuel assemblies and on microprobe measurements, which supply the information required to qualify the fuel inventory calculations.

5. Conclusion

The scope of the program which has been carried out is highly indicative of the interest industrialists are taking in the evolution of calculation techniques used to take burnup into account in criticality safety studies concerning transport, interim pool storage and dissolving of spent fuel assemblies. The objective for the coming years is to create a complete method containing all the expertise developed by the working group, with the aim of establishing a more fully optimised way of taking specific burnup into account in criticality safety studies, while demonstrating its conservative nature.

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(1) Courtois G. *et al.* Moret III. Un programme Monte Carlo pour le calcul rapide des coefficients de multiplication effectifs de milieux fissiles dans des géométries complexes. IPSN report 93/3

(2) Nuclear Recycling Recod'98. Conférence Recod'98 / Nice, 25 to 28 October 1998. SFEN/Recod'98

(3) R. Sanchez. 'APOLLO-II A modular code for multigroup transport calculations'. Nucl. Sci. Eng., 100, 352, (1988).

(4) Papers from the seminar held by SFEN and ADEPHYR entitled "Les Etudes de Criticité de Cycle du Combustible Nucléaire" chaired by M. Livolant. 29 January 1997 in Fontenay-aux-Roses, 30 January 1997 in Valduc.

(5) A. KAVENOKY, R.SANCHEZ
The APOLLO assembly spectrum code
Paris, April 1987

THE REBUS INTERNATIONAL PROGRAM

(CRITICAL EXPERIMENT WITH SPENT-FUEL FOR BURNUP-CREDIT VALIDATION)

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ABSTRACT

Actual trends in fuel management increase the fuel utilization by increasing the discharge burnup and irradiation times, resulting in an increase of the fuel initial fissile content.

On the other hand, some delay for final disposition of UO_2, and later for MOX-spent-fuel prescribes the storage of increasing quantities of spent fuel.

Burnup credit becomes a key issue in spent-fuel storage and shipment, allowing more compact racks for more enriched fuel (with costs reduction) than the classical approach which considers only fresh fuel for criticality safety evaluation.

As there is a recognized lack in experimental validation, BELGONUCLEAIRE and SCK•CEN have proposed to launch the REBUS International Program, consisting of reactivity measurements of well-characterized similar fresh and spent-fuel bundles in a zero-power critical facility.

This paper will describe in detail :

- Objectives.
- Test bundles design.

- Experimental procedure : Facility Description – Experimental Cores – Experimental Measurements and Techniques.
- Expected Δk (theoretical predetermination).
- Hot-cell work (Post-irradiation measurements before and after critical tests).
- Schedule.

The expected database will be used for burnup credit validation and licensing (including for derivation of usual safety margins).

1. OBJECTIVES

The current licensing approach for criticality safety evaluation considers only fresh fuel in every case. With the new trends, taking into account the burnup credit becomes a key issue for spent-fuel storage, allowing more compact racks for highly-enriched fuels than the classical approach. It will produce benefits in the reduction of necessary space and of the global cost as far as it will be accepted by Regulatory Authorities.

BELGONUCLEAIRE and SCK•CEN proposed to launch the REBUS International Program to validate at the international level the methods applied to answer to burnup credit issues. The LR-0 reactor located in Czech Republic and the VENUS Reactor located in Belgium are candidates for performing the experiments.

The spent-fuel bundles with about 20 fuel segments will have an active length of 1 meter, which is fully compatible with the critical facility height and some conditions of handling and transport. The fuel segments will be either fuel rods (1 meter long) irradiated in the BR3 PWR Reactor, for some tests, and for other tests, segments of the same length from PWR commercial spent-fuel elements (BWR commercial fuel elements can be considered in a next phase of the program) with very high burnup (50 to 60 GWd/tHM).

The critical tests for the determination of the burnup credit will be performed on fresh and spent fuels, by means of a differential measurement technique in order to avoid systematic uncertainties.

The present definition of the test program starts with the case of a reference bundle of absorber elements (without fuel) in order to check the handling and measurement procedures. Then, the selection of test bundles includes fresh UO_2 fuel and irradiated UO_2 fuel of the two following types :

↳ BR3, 5 % U235, medium burnup fuel with long cooling-time, representative of spent fuel to be stored for intermediate or final storage.

- Commercial PWR fuel (3.8 % U235) with high burnup and limited cooling time, representative of present fuel to be stored or transported.

Moreover, fresh and irradiated MOX fuels of the BR3 type (with 6.9 % Pu fissile) will be used in the test matrix. The possibility of testing irradiated BWR commercial fuel segments later on is considered as well in the feasibility studies.

Table 1 : REBUS PWR Test Matrix

		Fuel Type	Basic Description
RE-01	Reference absorber test bundle	(fuelless)	Use of B_4C
RE-02	Fresh UO2 (BR3)	PWR-UO_2	5 w/o U-235
RE-03	Irradiated (BR3)	PWR-UO_2	5 w/o U-235 ; 30 GWd/t
RE-04	Fresh MOX (BR3)	PWR-MOX	6.9 w/o Pu-fiss.
RE-05	Irradiated MOX (BR3)	PWR-MOX	6.9 w/o Pu-fiss, 20 GWd/t
RE-06	Fresh UO2	Commercial PWR UO_2	3.8 w/o U-235
RE-07	Irradiated UO2 – High burnup	Commercial PWR UO_2	3.8 w/o U-235 ; 60 GWd/t

Section 2 of this paper describes the test bundle design.

Section 3 deals with the experimental facility and the measurement procedure in zero-power reactor.

Preliminary evaluations of the expected reactivity effects (Δk) due to burnup, for a typical PWR fuel are given in Section 4. A summary of the REBUS data base as well as the expected planning is given respectively in Section 5 and 6.

2. TEST BUNDLE DESIGN AND FABRICATION

The design for PWR tests refers basically to a square assembly of 5 x 5 rods, with the same pitch as in PWR fuel assembly, constituted by about 20 to 25 fuel rods or fuel segments of one meter length.

A good neutron moderator material is located between the rods of the bundle ; it can be light water or an hydrocarbonaceous polymer block. In case of water moderator, the design of the bundle would be similar to a PWR assembly with guide tubes and grids. Another option is to choose a tight block of polythene to contain the bundle. Polythene $(CH_2)_n$ is commonly used in zero-power reactors ; its specific mass is close to that one of water and its hydrogen mass ratio is lonely a little higher than for water.

The bundle design in a uniform polythene block represents a rather simple fuel rod assembly without grids with a pitch similar as in a commercial reactor ; 100 mm of polythen is added at the top and at the bottom of the rods to close the bundle, simulating also the effect of reflectors. This design can easily be adapted to the dimensions of BWR rods.

Some irradiation resistance tests were performed at the SCK•CEN (Mol) for the polythene in order to check the behaviour of the material under the high flux of gamma rays coming from the burnt fuel rods. The integrated dose due to the stay of the irradiated rods in the bundle during several weeks could be a few MGy in the worst case (60 MWd/kg HM burnup plus 2 years of cooling time). Samples were irradiated in the BRIGITTE facility of the BR2 reactor up to 10 MGy of Co60 gamma-rays, at a high dose rate of 1 MGy/day.

Controls of the mechanical properties show modifications of the stress-elongation curve probably due to polymerisation and cross links of the CH_2 chains during irradiation, but the residual ductility remains high enough for the use of polythene as structure material of the bundle in a zero-power reactor. On the other hand, the loss of hydrogen during the gamma-ray irradiation was found negligible, so that the moderation property of the polythene will not be affected.

Thermal analysis indicate that the residual power of the test bundle could be evacuated through the polythene block without bringing it to unacceptable temperature during the transport in a closed container submitted to a 38 °C outside temperature and the maximum sunshine heat flux imposed by regulations (400 W/m^2).

In order to avoid any contamination in the reactor water, the cladding of all the tested rods must be free of any sign of failure. Then, wet-sipping tests will be performed. Finally, an efficient decontamination process will be applied, for instance, the well-known Cerium sulphate process. After satisfactory decontamination, the rods will be transferred to hot-cells facility at SCK•CEN for additional non-destructive examinations in order to possibly evidence some non-visible clad defects.

The bundle assembly including the rods and sealed will be tested for global tightness before shipment in the transport container to the reactor building.

3. EXPERIMENTAL FACILITIES AND MEASUREMENTS TECHNIQUES

3.1. LR-0

The zero-power reactor LR-0, operated by the Nuclear Research Institute REZ, plc, is located in Rez, near Prague (Czech Republic). LR-0 is an experimental reactor for determination of the neutron-physical characteristics of VVER- and PWR-type reactor lattices and shielding with UO_2 or MOX fuel.

The basic data of the reactor are :

- Maximal thermal flux = 10^{13} neutrons/m^2.s.
- Atmospheric pressure.
- Fuel assemblies are shortened dismountable models of VVER-1000, with 312 pins in a triangular lattice and 18 absorbing element channels (stainless-steel tube) for boron carbinde control clusters and 1 central instrumentation tube (Zr).
- Control by the change of moderator level and control cluster position.
- Vessel dimensions : 3.5 m diameter, 6.5 m height.
- Twelve horizontal dry channels with control system chambers.

The fuel pins include sintered UO_2 pellets in a Zr cladding tube ; the total length is 1357 mm and the active length is 1250 mm.

The reactor vessel is made of aluminium ; it is located in a concrete reactor bunker and is covered with a mobile concrete shield. Within the reactor vessel, the core is supported by a plate accommodating the VVER-1000 type fuel assemblies in the triangular lattice with the pitch of 236 mm.

The power is controlled by the moderator level in the reactor core and/or by the system of absorption clusters. At least 6 out of all absorption clusters must be selected as emergency clusters and they are kept at the upper position during reactor operation. The other clusters serve for experimental purposes, one of them being movable during operation.

For the REBUS experiments, it is necessary to modify the facility in order to introduce the test bundle in the central assembly of the LR-0 core and to allow remote handling of the high burnup bundle. When the bundle is introduced in the reactor vessel, it is located inside a lead container in order to reduce the high gamma doses (Figure 1).

The core of LR-0, proposed for REBUS (Figure 2) will include a driving zone consisting of 6 VVER-1000 type fuel assemblies with 2 % U235 enrichment, and a speciel fuel assembly in the core centre, in which the test bundle will be introduced ; this central assembly will be especially prepared with a square lattice pitch and only 12 cluster tubes as structural elements so that this assembly is a continuous lattice of fresh LR-0 fuel rods and REBUS (fresh or irradiated) rods.

Before and after every criticality measurement, the lead container with the bundle will be placed in a special tube located between the bottom part of the vessel and the reactor bunker.

The criticality of the reactor in the REBUS tests is to be reached by adjusting only the moderator height in order to have cores without any perturbation. The critical conditions for a core loading is defined by the critical moderator level H_{cr}. The reactivity can be determined by inverse kinetics method as a function of the moderator height.

3.2. VENUS

The VENUS critical facility is a water-moderated zero-power reactor. It consists of an open (non-pressurized) stainless-steel cylindrical vessel including a set of grids which maintain fuel rods in a vertical position. The maximum thermal neutron flux can reach 5.10^9 n/cm^2.s and can be maintained for 10 hours/day.

Criticality is reached by raising the water level within the vessel : water is pumped to the vessel from a storage tank which contains water when the reactor is not in operation. Reactivity control is obtained by controlling the water level in the vessel or by means of absorbing rods if experiments are performed at a nominal water level. Since the neutron flux is very low, no water circulation is needed to keep the working temperature at ambient room conditions.

The VENUS critical facility can simulate the neutronic behaviour of PWR reactors as well as BWR reactors thanks to the compositions and dimensions of the available fuel (MOX with several Pu enrichments, UO_2) and absorber rods (B_4C, Ag-In-Cd, Gd_2O_3-UO_2), the grid pitch size (12.6 mm), the nature of the moderator, the water density reduction simulation rods – i.e. Al rods –, the easy loading feature (which allows the loading of many different core configurations), ... Therefore VENUS is used for the validation of computational tools (codes and cross-sections libraries) used for ex-core neutron transport and core physics calculations of MOX and UO_2 assemblies loaded in LWR reactors;

For the REBUS experiment, it is necessary to modify the VENUS facility in order to introduce the test bundles in the central assembly of the VENUS core and to allow remote handling of the high burnup bundles temporarily introduced into a lead container in order to reduce the gamma doses to the operators (Figure 1).

The core of VENUS, proposed for REBUS, will include a driving zone consisting of a square of 25 x 25 3.3 % U235 rods, surrounded by 2 rows of UO_2 4.0 % U235 rods, both of them surrounding a large hole in the core centre ; this central hole is designed to receive the test bundle when it is lowered from the lead container through the cone-shaped pierced metal plate ; the resulting LWR square pitch of the whole core, including the test bundle, is 1.26 cm overall.

Once a test bundle is loaded in the core, the reactor can be started (water can be pumped). Criticality is reached by adjusting either the water level or the control rods height. The resulting accuracy on the criticality is less than 25 pcm. The reactivity coefficient can be measured with an accuracy better than 3 %.

Beside the hereabove mentioned modifications, another modification has to be made : the in-vessel structures have to be adapted to receive 1 m long (active length) fuel rods instead of 50 cm long (active length) fuel rods. This modification is essential to ensure sufficient Δk between fresh and irradiated fuel.

4. THEORETICAL PREDETERMINATION

Preliminary evaluations of the burnup reactivity effects in LR-0 and in VENUS have been performed for different cases.

The WIMS-7 or the WIMS-8 code are used with the associated JEF 2.2 based library for simulating the irradiation and decay of the tested fuels.

Unit cell calculations considering the different fuels, and the driving zone in hexagonal or in square lattice geometry, have been performed ; the cross-section of the fuels are calculated in a 16-group structure as used in standard criticality safety evaluations.

The calculations of the experimental core have been performed with the DANTSYS code system in 2-dimensional geometry (TWODANT solver), assuming the core is totally flooded.

A simulation of the core layout in LR-0 and in VENUS is provided in Figure 2.

Examples of reactivity differences between fresh and irradiated bundles illustrated respectively by simulating the typical case of high-irradiated commercial fuel in LR-0 and in VENUS (Figure 3).

The scope of the REBUS Program will include several critical tests of PWR fuels essentially (UO_2 or MOX). The test matrix for PWR fuels is given in Table 1.

5. THE REBUS DATA BASE

The data base will consist of :

- Zero-power reactor characteristics.
- Fuel rod characteristics (fresh fuel parameters, isotope content, irradiation features, calculated densities).
- Hot-cell data before critical experiments such as gamma-scanning of rods and bundle characteristics.
- Critical experiment description and results for each test case.
- Post-irradiation examinations after critical experiments for some selected rods (radiochemical analysis to determine the burnup, the actinides and selected fission products) as well as measurement of the activation of flux wires initially installed in the test bundles.

6. PLANNING

The program startup is scheduled before end of 1999. The modification of the facility, the fuel procurement, the rods preparation and the bundle fabrication as well as the blanco test will be performed during the year 2000.

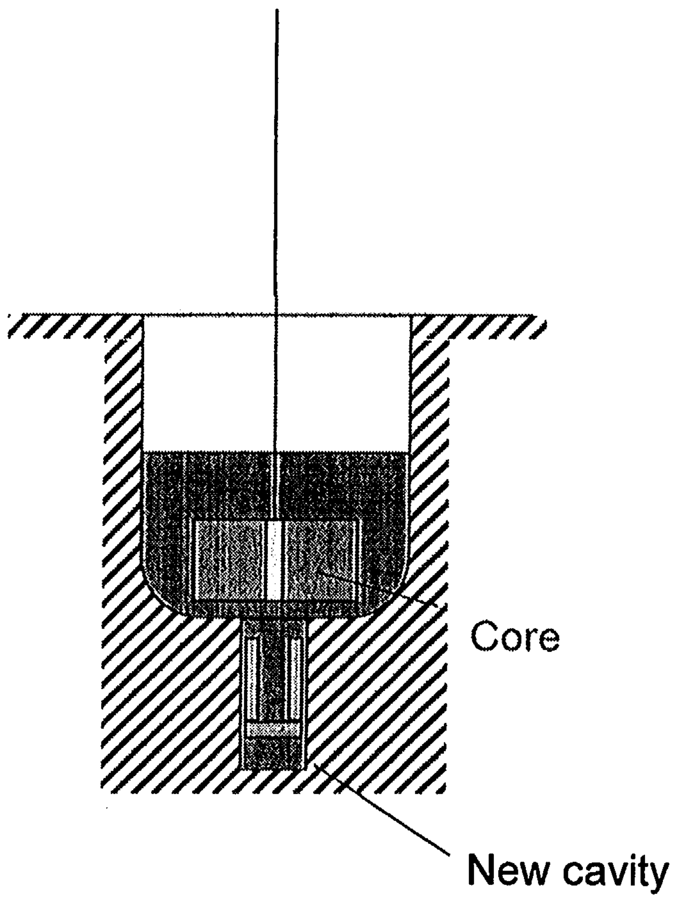
Shipment in test reactor, reactivity measurements and PIE analysis will be performed from early to mid-2001.

Data base analysis, reporting, etc ... will therefore be prepared between mid-2001 and mid-2002 with the target to deliver a Final Report at around mid-2002.

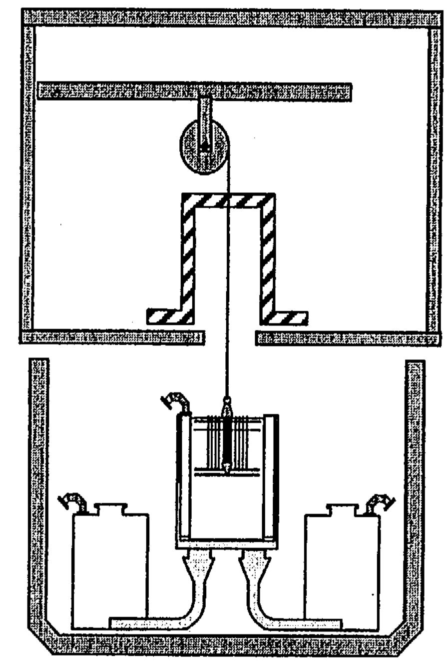
7. CONCLUSION

The REBUS International Program is designed and prepared to answer to present and future issues in nuclear fuel safety and fuel-cycle backend. It will allow to make advances in the burnup credit validation and application to numerous safety aspects, implying several R & D valuable tools and improving the international data bases on criticality aspects for UO_2 and MOX LWR fuels.

LR0 Experiments



VENUS Experiments

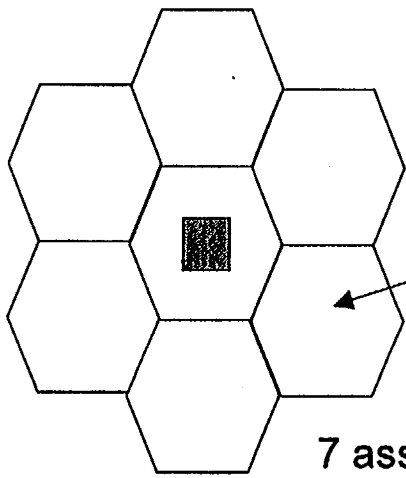


N.B. Bundle shielded by lead containers

Figure 1

Core lay-out

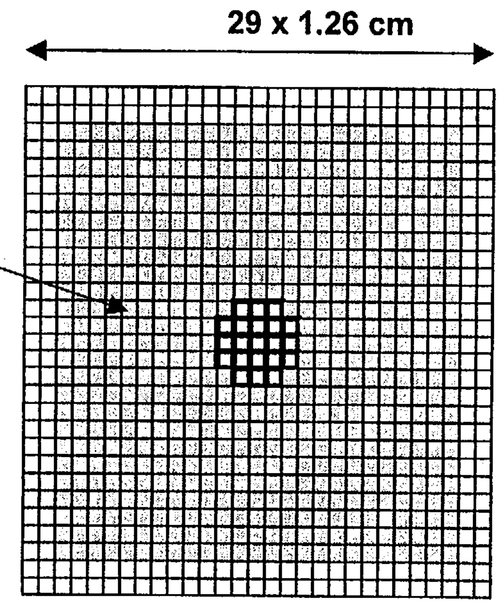
LRO



7 assembly configuration

Feeding zone

VENUS



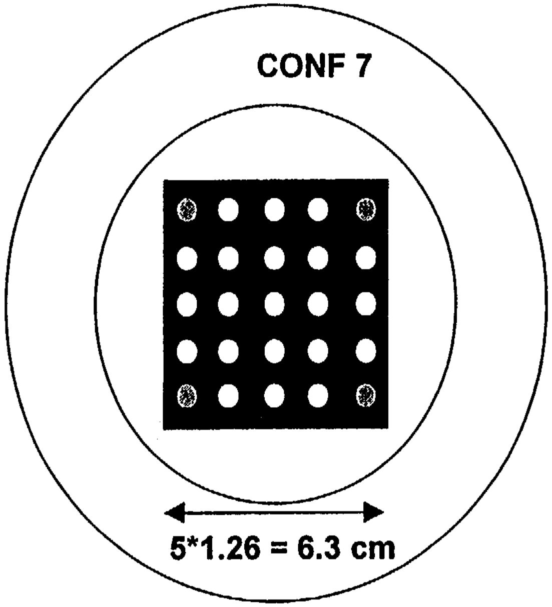
☐ **Test Bundle** ☐ **U 3.3** ☐ **U 4/0**

	UO_2 VVER Fuel	UO_2 BR3 PWR Fuel
Oxide density (gr/cc)	10.33	10.25
Pellet outer diameter (mm)	0.753	0.819
Rod outside diameter (mm)	0.915	0.950
Cladding material	Zr (99%)+Nb (1%)	Zircaloy 4
Cladding thickness (cm)	0.065	0.114
U-235 enrichment (w/o U-235/U)	2.0	3.3 / 4.0
Lattice pitch (cm)	1.275	1.260

Figure 2

Δk Fuel

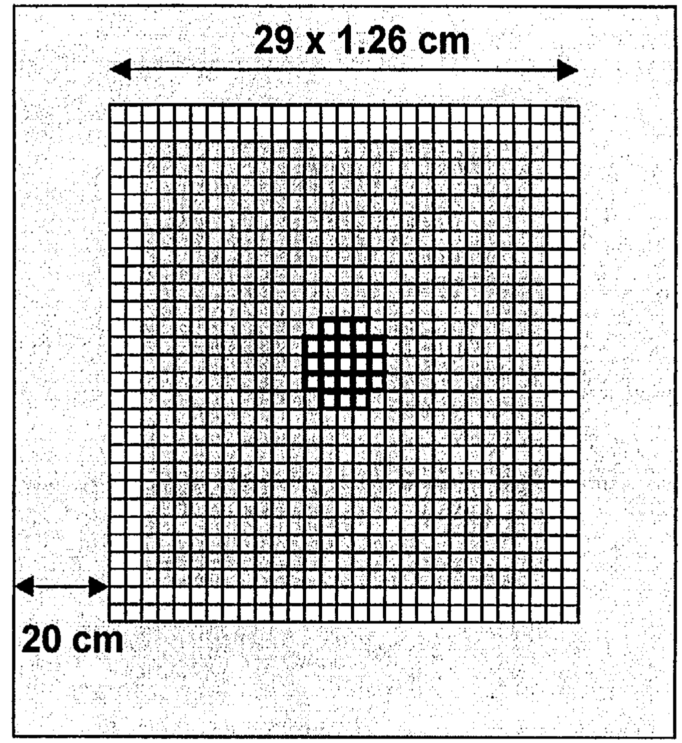
LRO



U 2.0% HEXA PA

■ Polyethylene (PE)

VENUS



☐ Test Bundle ☐ U 3.3 ☐ U 4/0 ☐ H_2O

BOX	CONF 7	Δk (pcm)
PE + 5x5	1.09516	
PE + 5x5	1.07803	-1576

Test Fuel	B.U. (rod avg.)
U 3.8%	fresh
U 3.8%	55 GWj/t + 3y

Rods	Box	k-eff WIMS	Δ-k (pcm)
21	H_2O	1.05225	
21	H_2O	1.03426	-1724

Figure 3

United States
Nuclear Regulatory Commission

IMPLEMENTATION OF THE REVISED SOURCE TERM AT U.S. OPERATING REACTORS

Presentation at the Water Reactor Safety Meeting

October 27, 1999

Jason H. Schaperow
Jay Y. Lee

Overview

NRC published revised source term (NUREG-1465)

Based on severe accident research

More realistic release rate of fission products

More realistic physical and chemical forms of fission products

Motivation: More realistic source term provides safety and cost benefits

Source term implementation effort consists of

(1) Rebaselining
(2) Pilot Plant Applications
(3) Rulemaking

Rebaselining

Objective was to develop a better understanding of the impacts of implementing the revised source term for operating reactors.

- Effect on calculation of individual offsite and control room doses

- Effect on calculation of dose for equipment qualification

- Effect of potential plant modifications, including severe accident risk impacts

Rebaselining was an assessment of the likely dominant issues as revealed by analysis of representative plants and served as "test bed" for developing regulatory criteria

Rebaselining results provided in SECY-98-154

Comparison of TID-14844 and NUREG-1465 Source Terms

TID-14844	NUREG-1465
Instantaneous release	Release over 1.8 hr (2 hr for BWR)
100% Noble Gases 50% Iodines (with 50% Plateout) 1% Solids	100% Noble Gases 40% Iodine (30% for BWR) 30% Cesium (25% for BWR) 5% Tellurium 2% Barium .02%-.2% Others

Iodine

TID-14844	NUREG-1465
91% inorganic vapor 4% organic vapor 5% aerosol	4.85% inorganic vapor .15% organic vapor 95% aerosol

Solids

TID-14844	NUREG-1465
Solids normally ignored for offsite dose calculation	Solids treated as aerosol

Rebaselining Conclusions (1)

Offsite and Control Room Doses

Impact of NUREG-1465 vs. TID-14844 is to generally produce lower calculated doses (up to an order of magnitude)

Extent of the reduction is influenced by several factors

a. Influence of safety features which are timing sensitive (e.g., SGTS, subatmospheric design)
b. Analysis assumptions used in SER and FSAR calculations

Use of updated dose conversion factors will, by itself, produce lower calculated doses

Many of the types of plant changes being contemplated could be made and doses would remain within acceptance limits

Rebaselining Conclusions (2)

Equipment Qualification Doses

Similar doses for equipment exposed to containment atmosphere.

Higher doses later in time for equipment exposed to sump water, due to higher cesium inventory in NUREG-1465 source term.

Margin

Analysis performed with MELCOR severe accident code indicated that offsite DBA doses still have substantial margin (a factor of 2 or greater) even though the NUREG-1465 dose may be well below the earlier TID dose.

Rebaselining Conclusions (3)

Risk

Potential plant changes being contemplated with the NUREG-1465 source term are not likely to have substantial risk impacts, because most of the systems being changed are not involved in risk significant sequences.

Bottom Line

No issues were identified that would prevent implementation of the revised source term at operating reactors.

RADTRAD Code

NRC code to evaluate offsite and control room doses using the revised source term

Used extensively in-house in rebaselining and evaluating pilot plant applications

Version 3.01 issued June 1999 (NUREG/CR-6604, Supplement 1)

- more flexibility in nodalizing containment, improved output content and format, easy to use graphical user interface

- a dozen domestic users (utilities and industry consultants) so far, several more expected by end of year

- Website: http://www.nrc.gov/RES/RADTRAD

Pilot Plant Applications of Revised Source Term

Pilot plant applications use the revised source term for safety enhancements and reduction in unnecessary burden.

Pilot Plant	Reactor	Containment	Proposed Changes
Perry (completed)	GE 6	Mark III	eliminate MSIV leakage control system, increase allowable MSIV leak rate, implement pH control for sump
Grand Gulf (completed)	GE 6	Mark III	reduced requirements for containment isolation timing
Indian Point 2 (under review)	Westinghouse Four-Loop	Large, Dry	eliminate containment recirculation filters (charcoal and HEPA)
Grand Gulf (expected)	GE 6	Mark III	increase containment and MSIV leak rates, increase secondary containment drawdown time, improve control room
Oyster Creek (on hold)	GE 2	Mark I	no change; additional control room safety assessment using NUREG-1465

Perry Pilot Plant Application

Eliminate MSIV leakage control system and increase allowable MSIV leak rate.

> Because the revised source term is mainly aerosol, Perry was able to demonstrate that offsite and control room doses are within regulatory limits without the MSIV leakage control system and with a higher MSIV leak rate.

Implement pH control for containment sump water.

> Safety enhancement resulting from the severe accident insight that iodine will not re-volatilize from containment water when pH is controlled to be above 7.

Perry Pilot Plant Application

RADTRAD code used by NRC for review of Perry Pilot Plant Application

Insights gained from review of Perry Pilot Plant Application

> Most of the offsite dose is from leakage of drywell atmosphere through the MSIV and not from containment leakage
>
> Important processes affecting dose:
>
> > flow between drywell and wetwell affects the amount of fission products in the drywell available for leakage
> >
> > degree of mixing in the main steam line (i.e., plug flow versus well-mixed) affects deposition in the main steam line

NRC issued License Amendment approving application in March 1999.

Grand Gulf Pilot Plant Application

Grand Gulf submitted BWR Owners' Group report estimating that start of gap release is no sooner than 121 seconds for a BWR LOCA.

Allows changes to systems such as containment isolation systems that were previously required to respond to an instantaneous source term (from TID-14844).

> NRC used SCDAP/RELAP5 and FRAPCON codes to confirm the generic time of 121 seconds.
>
> NRC also used TRAC code with Grand Gulf plant deck to provide additional confirmation.

NRC accepted BWR Owners' Group report in September 1999.

Indian Point 2 Pilot Plant Application

Eliminate containment recirculation filters

Containment recirculation filters are part of the containment fan cooler Engineered Safety Feature.

Containment recirculation filters previously required because of the large organic iodine fraction (10%) used with the TID-14844 source term.

The revised source term has an organic iodine fraction of .15%.

This change is both a reduction in unnecessary regulatory burden and a safety enhancement; charcoal filters pose some fire threat and removal of the filters is a simplification of containment fan cooler Engineered Safety Feature.

Indian Point 2 Pilot Plant Application

Containment fan coolers

> Because the revised source term is mainly aerosol, impact of this aerosol on fan cooler heat removal performance is being examined.

> Includes consideration of non-radioactive aerosol released during the in-vessel phase of a severe accident.

NRC review of Indian Point 2 Pilot Plant Application to be completed by March 2000.

Rulemaking

Rule allows voluntary implementation of revised source term

- site boundary dose limit of 25 rem TEDE for any two hours

- existing dose limits for TID source term are 300 rem thyroid and 25 rem whole body for the first two hours

Regulatory guide

- provides detailed guidance on how to calculate dose

- issues include scope of re-analysis for accidents other than LOCA and need for re-analysis of equipment qualification

Final rule and draft regulatory guide recently submitted to Commission

Equipment Qualification

Rebaselining: doses were calculated for containment atmosphere and sump using TID and NUREG-1465 source terms

Gamma and beta doses in containment atmosphere

> Similar doses between TID and NUREG-1465 source terms, because the dose is from noble gases and iodine

Gamma dose for equipment exposed to sump water

> Higher at later times for the NUREG-1465 source term, because of the large amount of cesium in the NUREG-1465 source term.
>
> TID-14844 includes 1% of the core inventory of cesium, NUREG-1465 includes 30% of the core inventory of cesium.

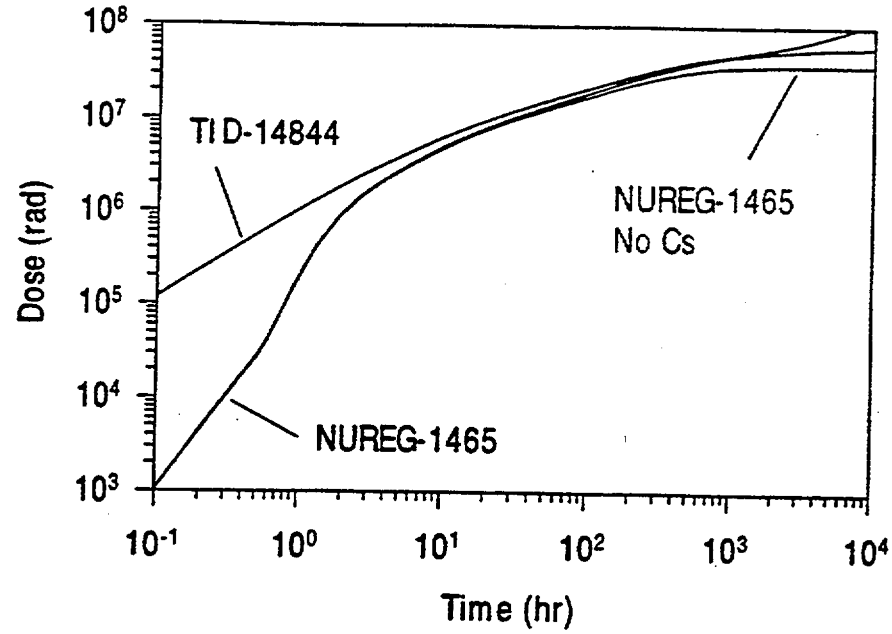


Figure 5-14. Surry Sump Doses.

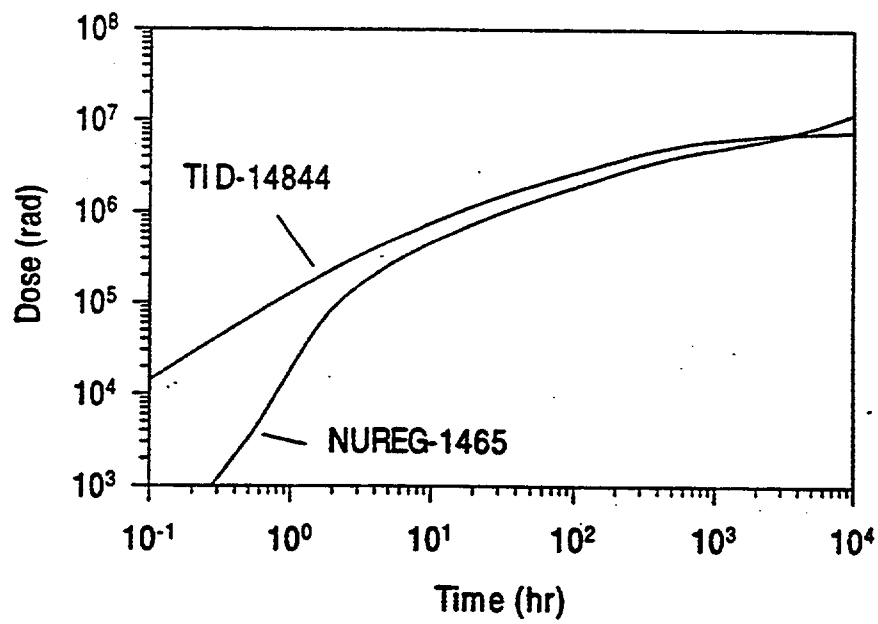


Figure 5-17. Grand Gulf Sump Doses.

Summary

Rebaselining study to evaluate impacts of using the revised source term completed.

Five pilot plant applications have been proposed; of these, two have been approved.

Expect numerous future applications using the revised source term.

The revised source term is a prime example of the benefits of severe accident research, namely, safety enhancements and reduction in unnecessary regulatory burden.

FIRST RESULTS OF THE LAST PHEBUS F.P TEST FPT4

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ABSTRACT

The Phebus FP program is a worldwide cooperative project designed to address issues associated with the fission product (FP) behavior in Light Water Reactors (LWR) under severe accident conditions. To date three experiments (out of six) have been carried out. The first two experiments, performed with a fuel rod bundle, provided information on FP releases up to fuel liquefaction, i.e. up to a relatively low temperature due to material interactions, and on FP behavior in LWR primary circuit and containment.

The last experiment FPT4, performed on July 22nd 1999, will fill the lack of understanding on the fission product release during the late phase of the core degradation. At this stage of the accident, the fuel rod can become rubblized forming a debris bed as observed in TMI-2. Therefore, the experiment was carried out with a prefabricated debris bed using fuel fragments and fully oxidized Zirconia shards. Five filters were located above the debris bed to collect the aerosols released.

The first analyses of temperature measurements as well as the non-destructive post test analyses show that the test was successfully performed regarding the fuel degradation objectives. The filters were operated according to a predetermined procedure for a "low aerosol release" scenario; further post test analyses and examinations will provide qualitative and quantitative results on the fission products and actinides trapped in the filters.

I. INTRODUCTION

The experimental PHEBUS FP (Fission Products) program is dedicated to the study of core meltdown in a light water reactor during a severe accident, and of the subsequent source term. It is more precisely devoted to the study of the FP release from fuel in degraded conditions (from core melt onset to pool formation), of the FP transport in the primary circuit of the reactor, as well as the FP behavior in the containment.

The program involves six in-pile tests [1] using fuel material and a scaled, well instrumented, primary circuit and containment models. Various thermal-hydraulic and physico-chemical conditions, typical of accident sequences, can be reproduced and the test fuel can be heated up to or beyond its melting point. The tests are carried out in the PHEBUS nuclear facility operated by the French Institut de Protection et de Sûreté Nucléaire (IPSN). The program is supported by Electricité de France (EdF), the European Union (EU), the USNRC (US), COG (Canada), NUPEC and JAERI (Japan), KAERI (South Korea), HSK and PSI (Switzerland).

Three experiments have been performed: FPT0 in December 1993, FPT1 in July 1996 and lately FPT4 at the end of July 1999. The main objective of this third test was to study the releases of low volatile fission products and transuranium elements from a rubble bed as well as the fuel vaporization. A secondary objective was to study the physical transition from debris bed to molten pool.

After a brief description of the test matrix and of the Phebus facility, the objectives and the experimental protocol of the test are presented. Then the very first results coming from the on-line instrumentation and from the non-destructive post test analyses are given.

II. THE TEST MATRIX

Before the FPT4 test, two other tests have been performed inside the PHEBUS FP program (FPT0 and FPT1). Both experiments were operated under similar thermal-hydraulic conditions, simulating a cold leg break under low pressure and steam rich environment. The main difference between the two tests was the burn-up of the test fuel. FPT0 used fresh fuel and FPT1 was carried out with a 23 GWd/tU irradiated fuel. The test fuel was a one-meter long bundle made up of 20 fuel rods and one absorber rod located in its center. Prior to the experimental heat-up sequence, the test bundles have been re-irradiated *in situ* a few days at nominal power, cooled by a forced flow of pressurized water. The objective of this re-irradiation was to re-create in the test fuel the inventory of short-lived fission products. The more striking findings of the source term are the presence of significant amounts of volatile iodine at the break of the cold leg and the insolubility of the iodine in the sump because of the silver released by the control rod.

The FPT4 test device was rather different from the two previous tests. The test fuel (33 GWd/tU irradiated) consists of a debris bed made of fuel fragments and fully oxidized Zircaloy cladding shards. It is the only test of the matrix using this type of fuel geometry.

The following test FPT2, is part of the FPT0-1-2 series, starting from fuel bundle geometry and including a silver-indium-cadmium control rod. It will be performed under steam poor conditions and boric acid will be introduced as an additive to the coolant flow, with a potential impact on fission product chemistry.

Two other tests are planned with fuel bundle geometry. In the FPT3 test, a boron carbide control rod will replace the silver-cadmium-indium control rod. The definition of the characteristics of the last test of the test matrix (FPT5) is still underway.

III. FPT4 TEST OBJECTIVES

The objective of this test was first to investigate the release of low volatile fission products and transuranium elements from a solid debris bed. This geometry, compared to "bundle" geometry promotes the releases of such a type of elements, because the particle specific area (ratio of the surface of the particle over its volume) of emission in contact with the steam is larger. In addition, the steam oxidizes the fuel, and this promotes the diffusion of the F.Ps in the fuel matrix.

At high temperature, it remains a large uncertainty on the rate of the fuel volatilization. This phenomenon is also promoting the emission of the fission products still in the fuel. At the time being, those types of releases have not been largely studied because they need to work at very high fuel temperatures (.~ 2430°C)

The second objective of the test was the study of the release during the transition from a solid debris bed to a molten pool and from the molten pool.

No re-irradiation of the test fuel was planned for this test, taking into account the geometry of the test fuel, which did not allow insuring its confinement and cooling during a re-iradiation phase. This led to a low I131 inventory; this was not judged as an important point for FPT4 because the objective of this test was the quantification of the released aerosols and not the chemistry of iodine in the containment. Therefore, it had been decided to collect the released elements in filters located in the experimental device, above the top of the debris bed, to be in the better conditions for the evaluation of the mass balances.

III. THE FPT4 TEST DEVICE AND THE PHEBUS FACILITY

THE PHEBUS FACILITY

The PHEBUS FP facility consists of:

- The PHEBUS driver core (pool type reactor with a maximum power of 40MW),
- A pressurized water loop used to cool the experimental fuel during the re-irradiation phase, and the experimental test device during the experimental phase,
- An injection loop which allows to reproduce the required thermal hydraulic conditions by supplying flows to the test train through three injection lines (steam, hydrogen, helium),
- A replica, on the 1/5000th scale, of the main elements of a nuclear reactor with a primary break on cold leg (called the « FP circuits ») including a U tube simulating a PWR steam generator and a tank (called REPF 502) simulating the containment of the reactor. Filters located inside the experimental test device (see after) collect the FPs. A by-pass line has been built specifically for the test FPT4, between the outlet of the experimental device and the REPF 502 (see figure 1). This line avoids contamination of the circuits of the next test. It allows to drive the gases flowing out of the filters to the REPF 502 tank where the vapor injected in the debris bed is condensed and the injected hydrogen and the fission gases released are collected.

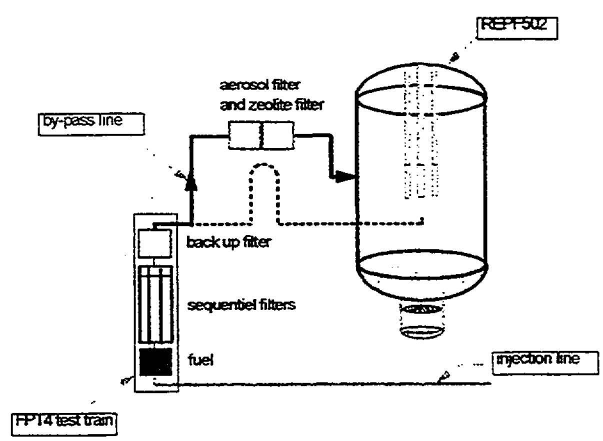

Figure 1 : Circuits and filtration device

THE FPT4 TEST DEVICE (figure 2)

It consisted of the fueled test section and the filtration device. It was located inside the test cell at the center of the PHEBUS reactor, which provides neutron flux to heat the test fuel by fission.

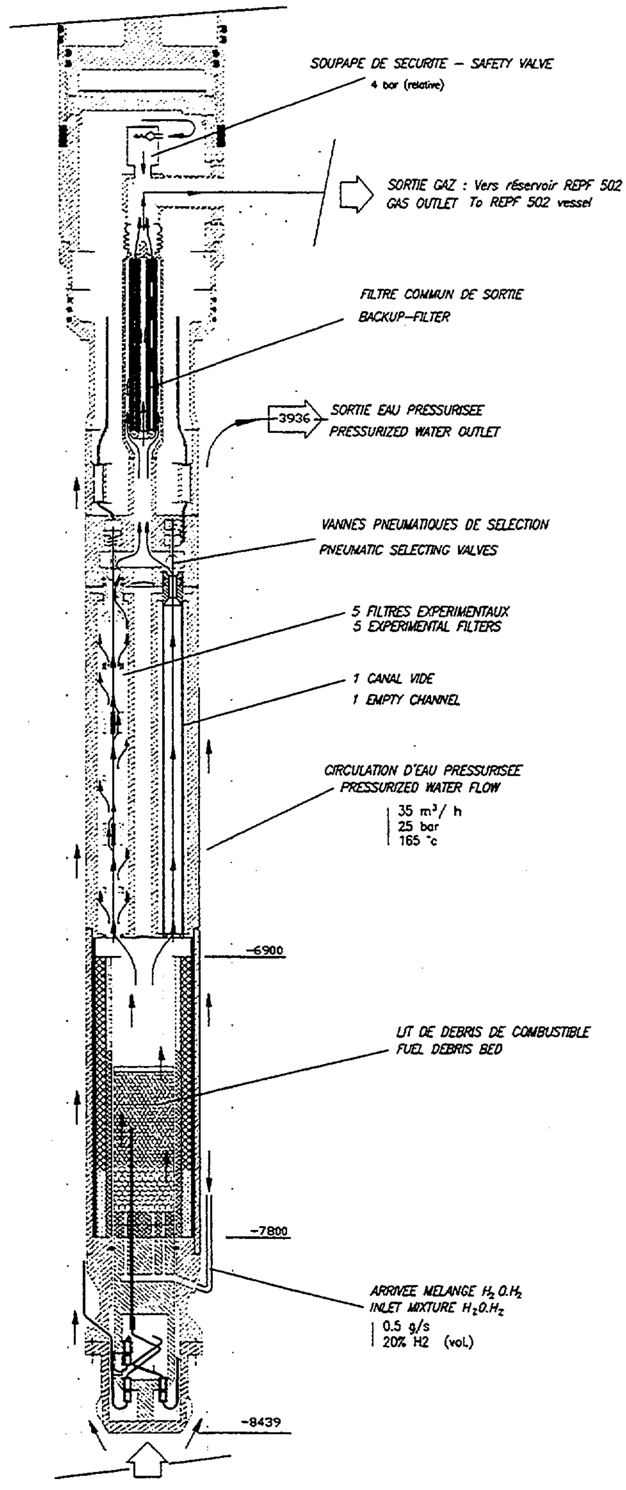

Figure 2 : FPT4 test section

- **The fueled test section** (figure 3)

It included the debris bed surrounded by an insulating shroud (Thoria and Zirconia sleeves) which insured the thermal insulation of the debris bed and protects the experimental device from the molten materials. The upper part of the debris bed named « active bed » was composed of PWR fuel pellet fragments with an average burn-up of 33 GWd/tU mixed with oxidized Zircaloy cladding shards. This « active bed » was 24 cm long, with a radius of 3.68 cm. It was sitting on a layer of depleted UO2 (0.23% ^{235}U enriched) called « passive bed ». This second part had the same radius and was 12 cm long. Its aim was to limit the downward axial progression of the molten materials in order to protect the lower part of the experimental test device.
The debris bed porosity was about 50%. It was kept in place by a canister made of Zircaloy. The canister was inserted in the insulating shroud.

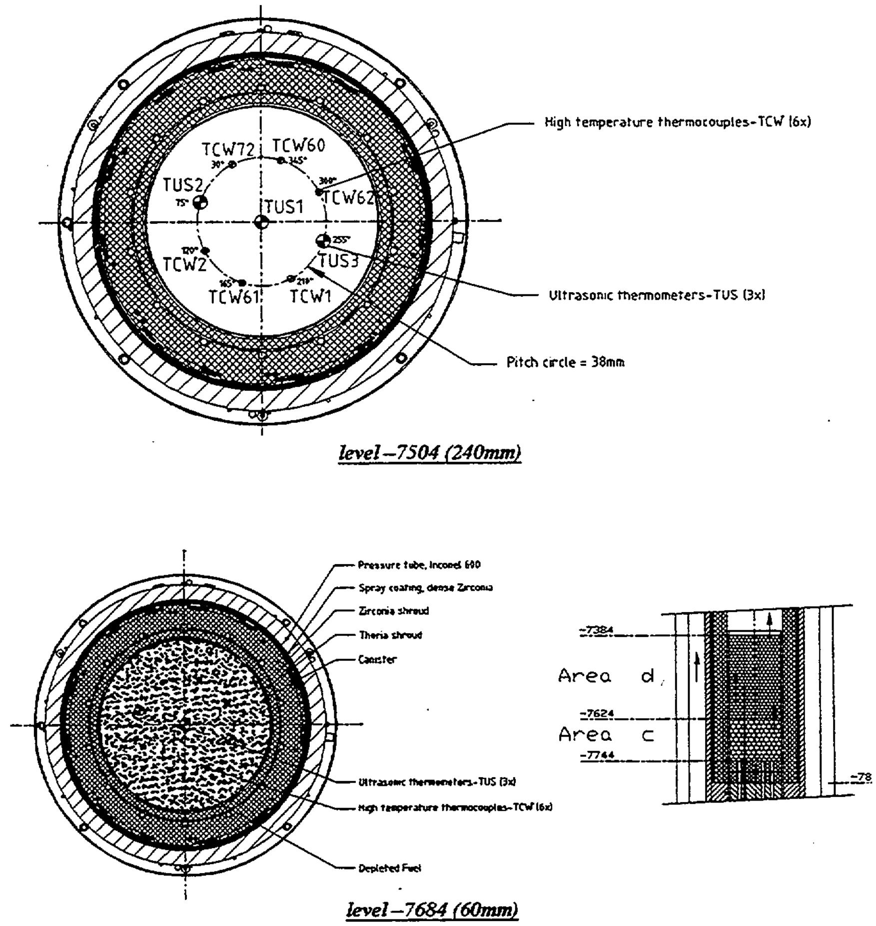

Figure 3 : Radial cross sections at debris bed elevations

The lower part of the « passive bed » was surrounded by an hafnia neutronic shield, which limited the nuclear power generated in this bed in order to prevent a too large axial melt progression during the pool phase.

The experimental test device was connected to two injection lines which allowed the injection of a steam – hydrogen mixture leading to physico-chemical conditions which are representative of the conditions encountered in a PWR in case of a severe core accident. The flow rate through the debris bed allowed to transport the aerosols from the debris bed up to a filtration device located in the experimental test device (see following section) just above the debris bed (see figure 2).

- **The filtration device** (figure 2 and 4)

A filtration system was settled in the upper part of the test train (55cm above the top of the debris bed), in order to trap the different aerosols released by the debris bed during the experimental phase and transported by the steam-hydrogen mixture. With that configuration the filtration system is very close to the source term.

This filtration system was divided into two filtration stages (see figure 4).
The **first filtration stage** (160cm long) was composed of 6 channels. Five of them were equipped with filters. Each filter was composed of 6 standard cartridges and 2 aerosols cartridges. The standard cartridges will be analyzed in order to determine the mass of the elements collected in the filters. The

aerosols cartridges were equipped with « deposition coupons » ; the analysis of the deposition on these plane plates will lead to the physical characterization of the aerosols (morphology, granulometry).
The last channel was a void channel;. Each channel could be opened sequentially thanks an opening/closing valve. One of these valves was a safety valve; it could open at a pressure threshold.
This system was designed to collect the released aerosols in one specific filter of the first stage for each characteristic phase of the test

At the outlet, the 6 channels gathered before entering the **second filtration stage** (70 cm long): the back-up filter. So, when a valve was opened, the gases went through the selected sequential filter, then through the back-up filter and finally the by-pass line toward the REPF 502 (containment simulator). The void channel placed in the first stage of filtration allowed to be able to use it in case of filter clogging and by-pass those filter

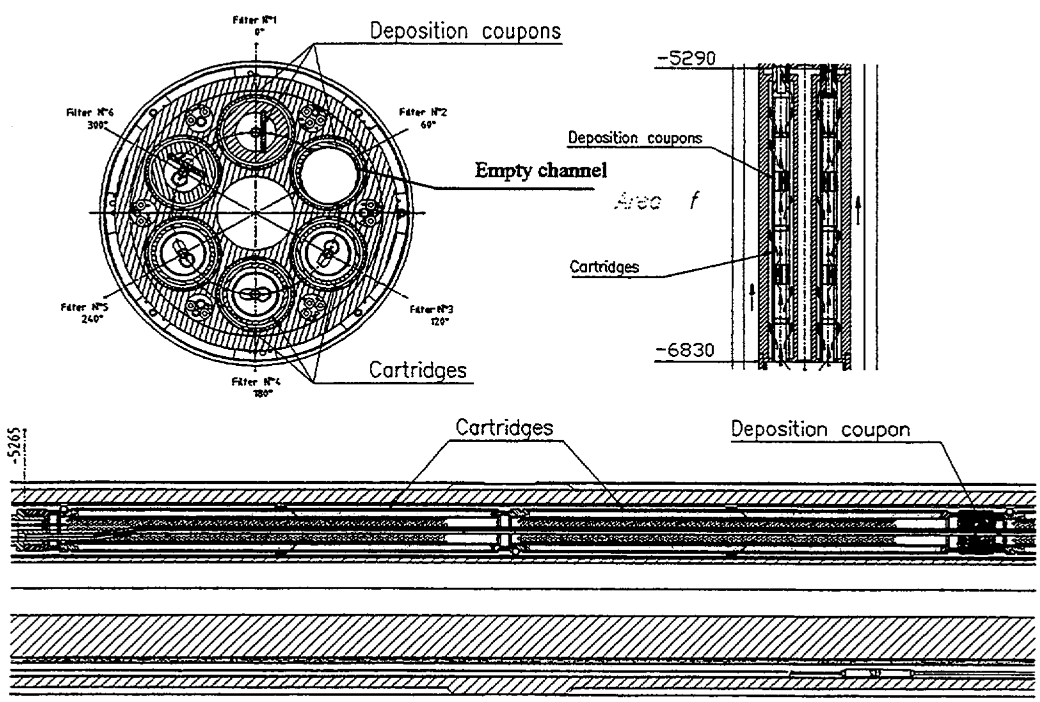

Figure 4 : First Filtration

Due to its specificity (fuel geometry, aerosols collecting system located inside the test device) the development of the FPT4 in-pile test device was a huge challenge. It has required a good deal of technological research, particularly for the elaboration of the final design of the filters and valves. A large amount of tests have been performed at PSI (Switzerland) in order to qualify this filter/valve system regarding the release objectives of the test.

IV. MAIN INSTRUMENTATION

INSTRUMENTATION OF THE DEBRIS BED AND OF THE SHROUD (figure 5)

The debris bed was instrumented with 6 thermocouples and 3 ultrasonic thermometers. Each of those thermometers would supply the temperatures on 7 different axial levels of the debris bed.

The shroud was instrumented with three sets of thermocouples:

- On the inner face of the ZrO2 shroud, called « Thoria-Zirconia » thermocouples
- Inside the Zirconia sleeves, named « Inside Zirconia » thermocouples.
- On the outer face of the Zirconia shroud, called « Outer Zirconia » thermocouples.
-

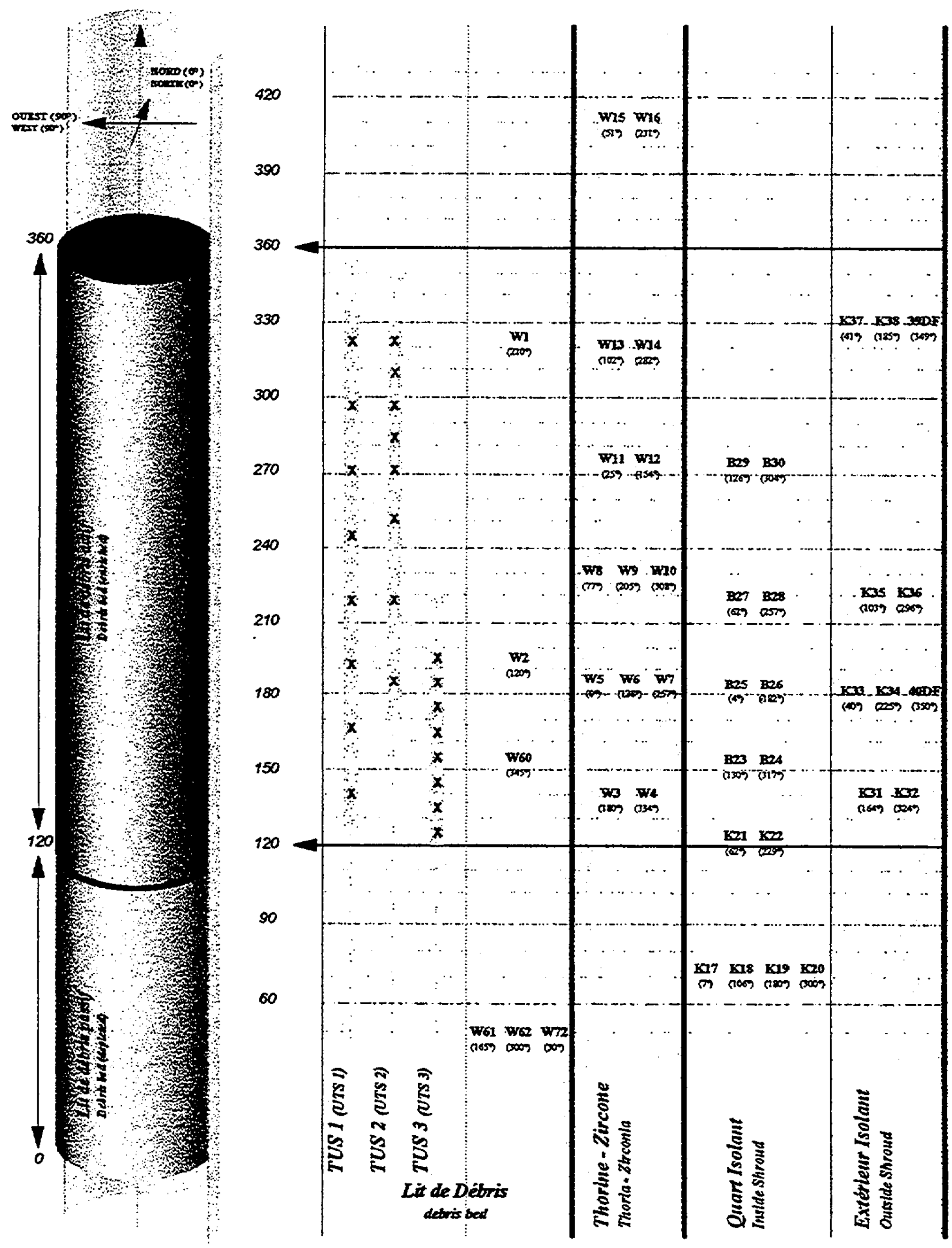

Figure 5 : FPT4 debris bed and shroud instrumentation

INSTRUMENTATION OF THE FILTRATION DEVICE

Eleven thermocouples measured the temperature at the inlet and outlet of the filters

Three pressure sensors (P3, P4, and P5) gave the value of the absolute pressure of the plenum, upstream the filtration device.
Two sensors (P1, P2) measured the pressure drop of the filtration device, this value being representative of the mass of the aerosols collected in the filters.
Three sensors measured the absolute pressure in the by-pass line.
Four on-line gamma spectrometry devices have been used to measure the activity in the circuit at the outlet of the test device, and in the REPF 502 tank.
Other information were available during the test, using the instrumentation located in the different systems of the facility such as the core power, the steam flow rate and the hydrogen flow rate.

V. THE EXPERIMENTAL PHASE (figure 6)

After reaching the nominal flow conditions through the bed, and following the test protocol, the test package was subjected to neutronic heating in the Phebus reactor which consists of a set of alternating transient and steady-state periods designed to take the debris bed in steps up to its melting temperature to form a molten pool. The five filters were operated sequentially during the different phases of the transient as specified.

This phase began the 22nd of July after about two weeks of preliminary phases. Its total duration was about 4h 30 min. It consisted of 4 phases:

- **The « thermal calibration » phase** which allowed to check the response of the instrumentation and the thermal properties of the FPT4 test device by varying the power dissipated in the debris bed, varying the steam flow rate, introducing the hydrogen injection. Four temperature plateaus have been carried out during this phase at 640°C, 800°C, 875°C and 1455°C in the fuel. It corresponds to reactor power levels of 0.31, 0.44, 0.44 and 1.44 MW respectively (P1 to P4 plateaus; see figure 6). The steam flow rate was 0.49 g/s except for P3 (0.2 g/s) and the hydrogen was injected at the beginning of the fourth plateau with a flow rate of 0.014 g/s.
  The time duration for this sequence was about 2 h 35 min.

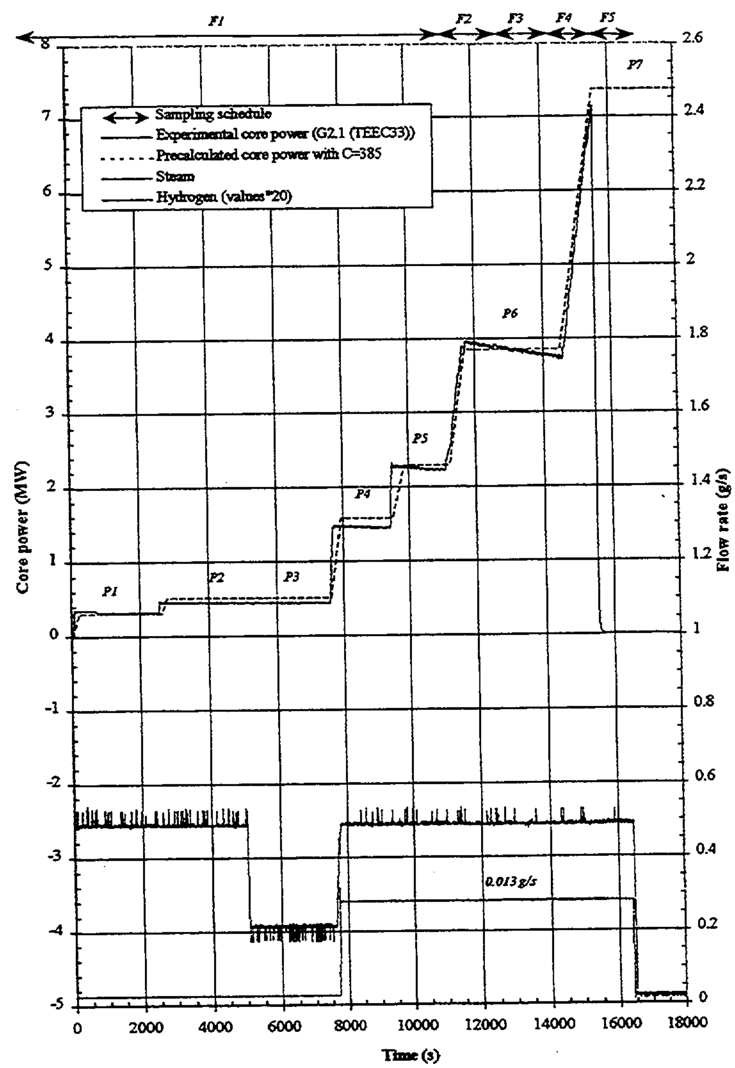

Figure 6 : Nominal power, steam and hydrogen injection flow rates and sampling schedule

- **The « release phase from a solid bed »** was devoted to collect the elements released from intact bed geometry. It consisted of two temperature plateaus at 1845°C and 2430°C. This last value is an estimation from the response of the instrumentation located in the upper part of the debris bed, which failed before the stabilization of the temperature at the temperature plateau. After this failure, the evolutions of the thermal behavior of the debris bed were monitored indirectly, using the thermocouples located in the insulating shroud.
  The associated reactor powers were 2.2 and 3.8 MW (P5 and P6 plateaus)
  The time duration for this sequence was about 1 h 25 min.

  The aerosols released during this phase were collected successively in three different filters (see figure 6):

  - the first filter (F1) was dedicated to collect the volatile fission products (essentially Cs),
  - the second one (F2) and the third one (F3) focused on the measurement of the aerosols emitted by the debris bed at very high temperature, but in intact geometry (« solid bed »). A volatilization of the low volatile fission products, and eventually of the Uranium oxide, was expected during this phase at very high temperature.

- **The « molten pool » phase**

According to the test protocol, this phase should have included a ramp from the last power plateau (P6) to a higher power plateau (P7) and stabilization at this power.
The maximal duration planned for this plateau was 2200s. The objective was to reach the liquefaction of the fuel and the formation of a molten pool during the ramp, then to stabilize the molten pool when stabilizing the power at P7. The released aerosols should have been collected during the P7 plateau

This phase was stopped shortly before the end of the ramp; the required value for the reactor power at P7 was 7.31 MW.. The power shutdown of the control rod of the reactor was carried out at a power level of 7.13 MW. Its time duration was 16 min.

The aerosols released by the fuel during the ramp have been collected in the F4 filter.

- **The cooling phase**

After the power shutdown, the cooling of the fuel was carried out by cooling the external part of the test device. The F5 filter was opened. The steam flow rate was kept to its nominal value during 15 min in order to cool the debris bed and to sweep the aerosols up to the filters.

V. FIRST RESULTS

During the P6 power plateau, the thermocouples located in the debris bed failed, as expected, at about 2200°C-2300°C. Then, the evolutions of the thermal behavior of the bed were monitored by means of the response of the thermocouples located in the insulating shroud (Thoria-Zirconia, Inside Zirconia, Outside Zirconia).

At the end of the P6 power plateau, one noticed an modification on the evolution of the temperatures pointing out a modification of the thermal behavior of the debris bed. The thermocouples located in the shroud and facing the upper part of the « active bed » (320mm) indicated a decrease of the temperatures in this region whereas the thermocouples in the shroud at lower locations (218mm, 234mm) showed an increase of the temperatures It was not foreseen and might be related to a motion of materials inside the debris bed. The events are now under analysis.

During the ramp from 3.7 MW (P6) to 7.13 MW, a significant displacement of the fuel has been detected, corresponding to the formation of a molten pool. It was detected by the deformation of the axial temperature profiles in the insulating shroud (see figure 7). This deformation is an accurate indication of fuel relocation. The fuel melts and relocates by gravity in the lower part of the debris bed, leading to an increase of the temperatures in the lower part, while the temperatures at higher locations decreases because of the loss of fuel which have moved down.

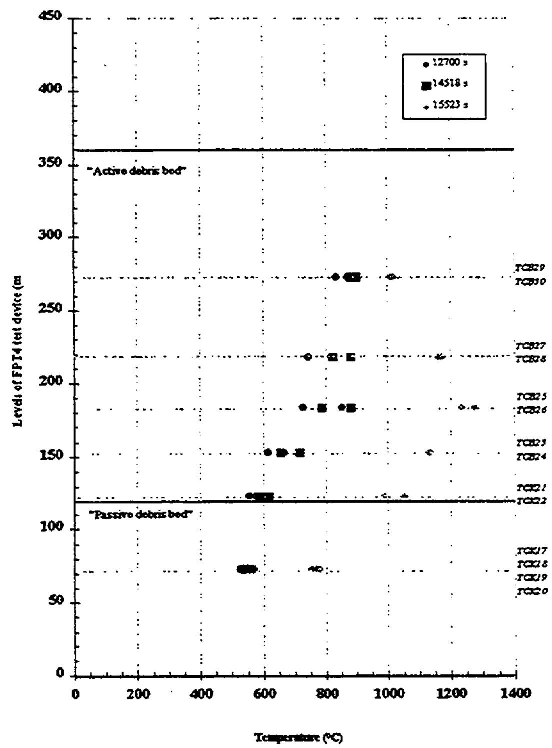

Figure 7 : Axial temperature profiles inside Zirconia

Those results were confirmed by the very first non-destructive post examinations (radiographies and gamma scannings). Comparing the initial and final state on the gamma scanning along the fueled test section (figure 8), one can notice a large axial region with a loss of fuel in the upper part of the debris bed, and an accumulation of fuel in the lower part. Some fuel seems to stay in place at its initial location at the top of the debris bed. The radiographies after the test (figure 8) at the location of the debris bed showed more precisely those three specific zones, indicating clearly that a molten pool was formed at the end of the test.

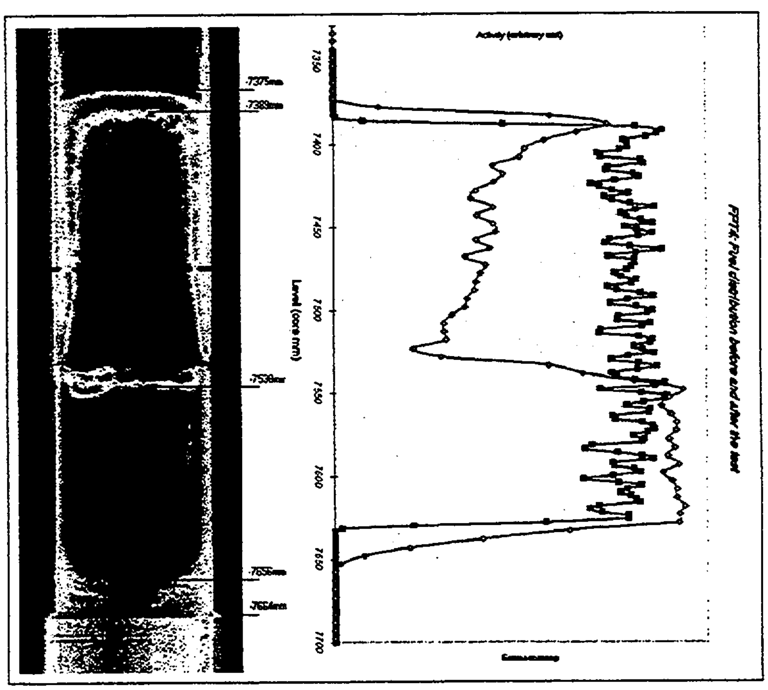

Figure 8 : Counting rates before and after the test along the debris bed, and radiography after the test

Concerning the mass and the type of the released aerosols, the first quantitative information will be supplied by non-destructive posttest analyses of the filtration device (gamma scanning and tomography). The first observations show that the filters present high activities when gamma scanned 24 hours after the test. Post test analysis and examinations are now under way.

During the « solid bed phase » at high temperature so as during the molten pool phase, the pressure drop of the sequential filters, resulting of their loading by the aerosols, never exceeded a few millibars for each filter. Taking into account the results of the qualification tests performed by PSI (Switzerland) on those filters, this tends to show that the loading of the filters by the aerosols is low. This has to be confirmed by the destructive post tests examinations.

The temperature of the steam-hydrogen mixture at the inlet of the F1 to F4 filters remained relatively low compared to what was precalculated. This might led to a significant re-deposition of the aerosols on the wall of the plenum.

VI. CONCLUSION

The FPT4 test was successfully performed regarding the fuel degradation and releases measurements objectives. The first non-destructive posttest examinations show that a molten pool was formed at the end of the test. The operation of the filters was as specified by a « low release » scenario and no filter blockage was observed. Posttest analysis on the filters in different laboratories will allow determining the mass and species of the collected aerosols for different levels of temperatures of the debris bed.

REFERENCES

[1] M. SCHWARZ, P. Von der HARDT
The Severe Accident Research Programme PHEBUS F.P :
First results and future tests, Proc. of the 23rd Water Reactor Safety Information Meeting, Vol. 2, P 239-261 – Bethesda, Maryland – October 23-25, 1995.

Insights on Application of the Alternative Source Term to Operating Plants

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1. Introduction

The source term, as used in the context of this paper, is the timing, magnitude, and form (chemical and physical) of the release of radioactivity from the reactor coolant system to the containment that would generally accompany a core damage accident at a nuclear power plant. A "standard" core damage source term has been part of the design and licensing basis for US Light Water Reactors (LWRs) since 10CFR100 and TID-14844 (References 1 and 2) were prepared and issued nearly 40 years ago. Advances in the understanding of core damage accidents since Three Mile Island (more than 20 years ago) have now resulted in a revision of the Design Basis Accident (DBA) source term.

This paper draws upon experience gained in having worked on the application of the revised DBA source term (Reference 3) (now called the Alternative Source Term, or AST, as applied to operating plants) to two Advanced Light Water Reactors (ALWRs) and then to the following operating plants:

- Browns Ferry
- Oyster Creek
- Perry
- Indian Point 2
- Grand Gulf
- Oconee

In addition, Polestar's relevant experience includes:

- Chairmanship of the Advanced Reactor Severe Accident Program (ARSAP) Source Term Expert Group (genesis of the AST)
- Preparer of the Electric Power Research Institute (EPRI)/Nuclear Energy Institute (NEI) Framework Document (Reference 4)
- Subcontractor to ABB-Combustion Engineering for application of the AST to the Korean Next Generation Reactor (KNGR)

The insights gained from involvement in all of these activities could be divided into areas of interest such as technical, regulatory, and economic. However, this is not a straightforward division because the insights gained from the application of the AST generally cut across all of these lines. Trying to effect such a division might obscure the reality that technically superior approaches may present regulatory challenges, and that such challenges can shape the economic viability of potential technical advances as much as (or even more than) the technical benefits to be gained. Therefore, the main body of this paper is divided into two main sections: (1) what is cost-effective now? (how can safety be improved with a return on investment?), and (2) what future advances are possible? (what are the research and development needs?).

2. What is Cost-Effective Now?

The AST (i.e., the release of radioactivity from the reactor coolant system to the containment) is largely a particulate source term. One may envision the post-core damage accident containment as being filled with a dense, radioactive smoke. While certain components of the source term (the krypton, xenon, and small fractions of the iodine isotopes) may be in gaseous form, these do not make up the dominant threat in terms of radiological potential.

2.1 Important Mechanisms for the Removal of Particulate

With a largely particulate source term (i.e., "smoke" or "dust"), trapping of the particulate in a variety of ways is likely. These would include the following:

- Sedimentation
- Heat transfer-related phoresis
- Impingement (e.g., on containment spray droplets or when converging towards very small leak paths at high velocity)
- Turbulent deposition
- Scrubbing

The effective size and density of the dispersed particulate generally affect these processes. As particle-to-particle collisions occur, the effective size or the particle increases by a process referred to as agglomeration. However, this process can also lead to porosity, which lowers the effective density of the agglomerated particle to below the theoretical value for the material. Nevertheless, agglomerated particles will settle more rapidly, and will also be more readily removed by impingement than will smaller particles. Sedimentation and spray removal are generally the two most important mechanisms for operating plants. A brief explanation of each of the five identified processes is provided in the following sections.

2.1.1 Removal by Sedimentation

Dispersed particulate will settle by gravity onto horizontal surfaces or surfaces with a horizontal component. The sedimentation removal rate is dependent on the particle size, effective density, and fall height, as well as on the density and viscosity of the atmosphere. Typical removal rates for sedimentation might be 0.5 to 1.0 per hour.

2.1.2 Heat Transfer-Related Phoresis

Whenever heat transfer is occurring from a gas to a surface, dispersed particulate will be swept to the surface by the difference in molecular energy associated with the temperature gradient (thermophoresis) or by the concentration gradient associated with condensation (diffusiophoresis). Such deposition can occur on spray droplets as well as on structural surfaces. Removal rates for phoretic deposition are generally comparable to sedimentation rates.

2.1.3 Impingement

When particulate is dispersed in a moving gas stream, and that stream encounters an obstruction causing the streamlines to either diverge (as in flow around an obstacle) or to converge (as in flow through an orifice), the particulate will cross the streamlines and impinge on the obstruction if the relative velocity is high enough, the particles are large enough, and the density is great enough. This is the dominant mechanism in the removal of particulate by sprays. The spray droplet acts as the obstruction with a relative velocity of several meters per second relative to the containment atmosphere. The larger the spray droplet's cross-sectional area, the larger the path "swept out" by the droplet as it moves through the containment atmosphere. Unfortunately, the water in the droplet increases with the cube of the diameter while the cross-sectional area increases only with square of the diameter. Therefore, smaller droplets make more effective use of a given spray flow rate. Typical spray removal rates would be five to 10 per hour, an order of magnitude greater than that for sedimentation or phoretic deposition.

One aspect of impingement is that a particulate-laden gas flow at a high velocity moving towards a small leak (e.g., a characteristic dimension of the order of a millimeter for a BWR main steam isolation valve leaking at its current limit) will almost certainly impinge at the entrance to the leak. This fact was well documented as part of the IDCOR Program (Reference 5, Issue 13a). In fact, in many cases the deposition is so effective that that leak path will be plugged by the particulate.

2.1.4 Turbulent Deposition

When flow is highly turbulent, mixing across streamlines is very high as well. The opportunity exists for particulate to come in contact with the boundaries of the conduit in which the flow exists (e.g., within a pipe) and remain at the wall. This is referred to as turbulent deposition. Removal rates for this process can be very high, and increasing the flow rate (and the degree of turbulence) can actually enhance removal even though residence time is decreased.

2.1.5 Scrubbing

When a particulate-laden gas flow is introduced below the surface of a liquid (and bubbles are formed), there is circulation within the bubble as the bubble moves toward the surface. This circulation, as well as heat and/or mass transfer when it exists can result in the transfer of particulate from the bubble to the liquid. Such behavior can, in turn, result in very effective decontamination of the gas stream. Decontamination factors of 10 to 100 are typical.

2.2 Impact of a Particulate Source Term on Plant Design and Analysis

For relatively leak-tight containments (i.e., containments which have not undergone a failure) and especially for containments with active removal systems such as sprays, the actual release to environment would be essentially only the very small gaseous component of the source term (i.e., noble gas, organic iodine, and gaseous iodine evolved from leaked coolant outside containment). The largest dose impact from this gaseous release would most likely be that of the noble gas. Plant designs (and supporting analyses) should reflect this reality. For example, filters in secondary containments may be relatively less effective than anticipated in current analyses.

In considering which systems, structures, and components (SSCs) or other elements of the plant configuration (e.g., operating procedures) should be looked at in terms of their real contribution to safety and cost-effectiveness in light of the AST, basically any SSC between the source and the offsite or control room dose is fair game. One should ask, "What performance, test, inspection, or maintenance requirements are the most onerous"? One may consider charcoal filter testing, testing/maintenance of MSIVs and/or MSIV leakage control systems (LCS), meeting engineered safety feature (ESF) recirculation leakage limits, and/or meeting control room unfiltered inleakage limits as examples.

In beginning a project to apply the AST, non-QA studies are a good way to start. It may be advantageous to consult all potentially affected groups within the plant operator's organization to compile and to assess a "wish list" of possible plant modifications. Then one can determine sensitivities and quantify trade-offs (including consideration of licensing risks and possible discussions with NRC). Once the non-QA work is complete and an understanding of what may be accomplished is achieved, then one may finalize the plan for the QA work and execute that plan.

In the course of the non-QA study and in preparing the plan one should consider innovative approaches in how the AST is applied. Only a very few applications have thus far been attempted, and even fewer have undergone review by NRC. There are still phenomena that have not yet been fully explored or developed in terms of application to licensing calculations. While licensing calculations should be done with a conservative bias, important phenomena should not be ignored in such a way that the results are qualitatively different from what one would expect in reality.

Applications to date have dealt with:

- Increased allowable containment, secondary containment bypass, MSIV, and ESF leakage
- Removal of MSIV-LCS
- Control room habitability issues (acceptable unfiltered inleakage and manual vs. automatic start of emergency systems or system modes)
- Removal or downgrading of charcoal in Safety-Related filter trains
- Extended closure times (or manual closure) for certain containment isolation valves
- Extended drawdown time for secondary containment

Associated NRC guidance (DG-1081) is developing rapidly. However, flexibility is still needed because of the limited number of applications and reviews. Some example comments on DG-1081 might include:

- Drywell-to-wetwell flow during assumed core degradation should apply to all BWR containment types, not just to Mark IIIs.
- The chemical form of iodine in the gap should be stated, thus giving applicants the option of assessing changes during transport rather than having only the "final" form stated in the guidance.

3. Possible Future Advances

Through the applications of the AST which have been completed to date (most of them identified in Section 1.0), much has been learned. Other potential areas where further work may be useful in terms of application and review of AST would include the following:

- Particulate removal at the entrance to small leak paths with high Mach Number (see Section 2.1.3)
- Behavior of reactor coolant containing iodine and other dissolved solids when leaked into steam generators (particularly steam generators which are dry)
- Revolatilization of iodine from water pools as pH decreases or if a free surface is exposed to a "clean" gas with a high exchange rate (in particular, the importance of iodate formation over time)
- Hygroscopicity (affinity for water) of cesium compounds (in particular, the impact on spray removal as airborne particulate becomes larger as the result of water condensation even in slightly superheated atmospheres - see Section 2.1.3)
- Dose conversion factors for organic iodine (as opposed to other iodine forms)

4. Summary

The process of applying the AST to operating plants is progressing well. The developing regulatory guidance contained in DG-1081 is welcome, but very prescriptive guidance at this juncture may be premature and even counterproductive. A balance must be reached and maintained between very prescriptive guidance and guidance which recognizes (and even encourages) innovation. NRC's approach to date has been very constructive.

5. References

1. Title 10 of the Code of Federal Regulations, Part 100

2. TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," U.S. AEC, Division of Licensing and Regulation, J. J. DiNunno, et al, March 23, 1962

3. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," February 1995, U.S. Nuclear Regulatory Commission.

4. EPRI TR-105909, Leaver, D.E.W. and Metcalf, J.E. *Generic Framework Document for Application of Revised Accident Source Terms to Operating Plants*. Interim Report. Electric Power Research Institute, Palo Alto, CA, 1995.

5. Technical Report 85.2, "Technical Support for Issue Resolution", Industry Degraded Core Rulemaking Program (IDCOR), July 1985

"Fission Product Release Under Severe Accidental Conditions; General Presentation of the Program and Synthesis of VERCORS 1 To 6 Results"

G. Ducros[1], P.P. Malgouyres[1], M. Kissane[2], D. Boulaud[3], M. Durin[4]

Abstract

The French Nuclear Protection and Safety Institute (IPSN) launched the HEVA-VERCORS program in 1983, in collaboration with Electricité de France (EDF). This program is devoted to the source term of fission products (FP) released from PWR fuel samples during a sequence representative of a severe accident. The analytical experiments are conducted in a shielded hot cell of the LAMA facility of the Grenoble center of CEA (Commissariat à l'Energie Atomique) ; as simplified tests addressing a limited number of phenomena, they give results complementary to those of the more global in-pile PHEBUS experiments.

Six VERCORS tests have been conducted from 1989 to 1994 with higher fuel temperatures (up to 2600 K) compared to the earlier HEVA tests [1] in order, in particular, to quantify better the release of lower volatile FPs. This paper gives an overview of the experimental facility, a synthesis of FP release from these tests and exhibits, as an example, some specific results of the VERCORS 6 test, performed with high burn-up fuel (60 GWd/tU).

The on-going VERCORS HT-RT program, designed to reach fuel liquefaction temperatures, is described before conclusions are drawn.

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1. Introduction

Because of the potentially severe consequences of a nuclear accident in terms of radiobiological effects, the international safety authorities initiated several experimental programs after the TMI-2 accident in order to improve the understanding and the mitigation of these situations. In France, the Nuclear Protection and Safety Institute (IPSN) launched the HEVA-VERCORS program in 1983, in collaboration with Electricité de France (EDF). This program is devoted to the source term of fission products (FP) released from PWR fuel samples during a sequence representative of a severe accident. The analytical experiments are conducted in a shielded hot cell of the LAMA facility of the Grenoble center of CEA (Commissariat à l'Energie Atomique) ; as simplified tests addressing a limited number of phenomena, they give results complementary to those of the more global in-pile PHEBUS experiments.

Six VERCORS tests have been conducted from 1989 to 1994 with higher fuel temperatures (up to 2600 K) compared to the earlier HEVA tests [1] in order, in particular, to quantify better the release of lower volatile FPs. This paper gives an overview of the experimental facility, a synthesis of FP release from these tests and exhibits, as an example, some specific results of the VERCORS 6 test, performed with high burn-up fuel (60 GWd/tU).

The on-going VERCORS HT-RT program, designed to reach fuel liquefaction temperatures, is described before conclusions are drawn.

2. Experimental apparatus ([2], [3])

Conducted with irradiated PWR fuel samples in a shielded hot cell, the tests are aimed at characterizing :

- the release kinetics and the total release of FPs, actinides and structural materials as a function of fuel temperature and oxidizing/reducing conditions of the environment,
- the aerosol source as a function of temperature,
- the chemical behavior of the FPs in the gas phase.

2.1. Fuel sample

The test fuel sample is a fuel rod section taken from a nuclear power reactor operated by EDF ; it is composed of three irradiated pellets in their original cladding. Two half-pellets of depleted uranium oxide are placed at either end of the sample and held in place by crimping the cladding (figure 1). Thus the cladding is not fully sealed.

The fuel sample is re-irradiated at low linear power (< 20 W/cm) in the SILOE experimental reactor for around seven days in order to recreate the short half-life FPs without inducing any in-pile release. These short half-life FPs, important for their radiobiological effects, include volatile FPs (iodine and tellurium), gas (xenon) and less volatile FPs (molybdenum, barium, ruthenium, cerium, lanthanum, zirconium ...). Since the experimental sequence is performed less than 40 hours after the end of the re-irradiation, direct measurement of ^{99}Mo, ^{132}Te, ^{133}I, ^{135}Xe, and sometimes ^{91}Sr is possible, using gamma spectrometry.

2.2. Hot cell lay out

The experimental apparatus is shown schematically in figure 2.

Along the path of gas flow, the following items are found :

- The main steam and hydrogen injection system,
- The system supplying helium gas, which is used to protect the graphite or tungsten susceptor from oxidization by steam,
- A superheater to heat the fluid flow up to 1100 K,
- The induction furnace itself to heat the fuel up to 2600 K. *It comprises, from the inside to the outside, two concentric channels dynamically sealed by a stack of dense zirconia sleeves (the internal channel containing the sample receives the steam and hydrogen flow, the external channel containing the susceptor (graphite or tungsten) is protected by the helium flow with a slightly higher pressure than the internal channel), a double-layer heat insulator (dense zirconia and alumina), a quartz tube, constituting the furnace chamber, and the furnace coils,*
- A junction zone, with a zirconia tube internal channel, linking the furnace to the impactor,
- A modified Andersen cascade impactor, with five stages, two granular beds and a back-up filter, used to trap the aerosols according to their size, is located in a resistive furnace with an adjustable temperature range from 500 K to 1000 K,
- A filter, heated at 400 K, which traps non-gaseous forms of iodine at this temperature, in place from the VERCORS 4 test onwards,
- A condenser and two dryers (silica gel and molecular sieve) for recovering the steam,
- A gas capacity to act as a buffer volume for on-line gas gamma spectrometry measurements, in place from the VERCORS 2 test onwards,
- A cold trap (charcoal adsorber cooled by liquid nitrogen) to collect noble gases.

The furnace provides a relatively flat temperature profile along the sample (< 50 K of gradient), thus insuring similar FP release for the three irradiated pellets in the fuel sample.

2.3. On-line instrumentation and post-test analyses

In addition to the conventional instrumentation (flow meters, pressure and temperature sensors), specialized on-line equipment is used :

- An optical pyrometer, calibrated between 1300 K and 3300 K, for measuring the temperature of the crucible base, which was used from test VERCORS 3 onwards,
- An on-line gas chromatography system located between the two dryers to measure the hydrogen emission kinetics during the cladding oxidization phase. This device is also able to quantify extraneous CO, resulting from partial graphite oxidization, when used as the susceptor material. This was in place from test VERCORS 2 onwards,
- Three complementary gamma spectrometers measure on-line FP release kinetics :
  - one detector is focused on the fuel rod. This unit records the FPs leaving the fuel. It has a low detectability level, around 10% of initial inventory of each FP, given the differential aspect of the release rate measurement at this location and the gamma self-absorption changes during fuel degradation at high temperature. Nevertheless, it is very useful because it is the sole unit able to measure the release of each FP,
  - one detector is focused on the top of the impactor. The advantage of this measurement lies in a high detectability level due to direct measurement of the released fraction, often less than 1% of FP initial inventory. On the other hand, only the FPs that deposit here can be detected, generally the most volatile ones,
  - one detector dedicated to the measurement of noble gases, typically the isotopes of xenon (^{133}Xe, ^{133m}Xe, ^{135}Xe, created in the SILOE reactor) and krypton (^{85}Kr created in EDF nuclear power plant). This unit provides a very high sensitivity and excellent measurement dynamics (from 10^{-9} to 10^{-1} of initial inventory per minute). It was used from test VERCORS 2 onwards.

These three gamma spectrometry units are composed of a portable liquid nitrogen-cooled Ge (High Purity) detector and an electronic device for signal shaping and storage, fitted with a rack unit for correcting pile-up counting losses in order to obtain spectra at a high rate (up to 1 spectrum.min^{-1}). Compensation better than 5% is guaranteed up to 150 000 pulses.s^{-1}, the limit of the counting rate defined for the tests.

After the test, the fuel is embedded in situ in an epoxy resin and X-rayed. A longitudinal gamma-scan of the fuel is conducted to measure the final FP inventory in order to calculate the quantitative fractions of FPs emitted by the fuel during the test. All the components of the loop (impactor stages, filters, condenser, dryers, etc.) are then gamma-scanned to measure and locate the FP released during the test and to draw up a mass balance of these FPs. For some tests, a nondestructive transversal gamma-scan is carried out for several angles of incidence to determine the spatial location of the FPs remaining inside the fuel and analyze possible interactions of these FPs with cladding components (for instance tellurium or barium trapped in the cladding, masking their emission) or their location inside the corium in case of fuel melting [4].

A ceramographic examination is carried out on each pellet of the fuel rod to analyze the changes in the microstructure of the cladding and the fuel.

Physico-chemical analyses are then carried out on samples of the loop after dismantling, especially on the impactor plates and on the zirconia sleeves linking the furnace to the impactor. The basic method used is scanning electron microscopy combined with analysis of X-ray emission (SEM/EDS), performed in the LAMA laboratory and in collaboration with the AEA-Winfrith laboratories. Some XPS and XRD analyses have been conducted on selected samples.

3. FP release synthesis

3.1. Test matrix parameters

The parameters that can be changed are the temperature plateau, the temperature ramp and the burn-up of the fuel sample, the temperature of the impactor, the fluid composition and flow rate (steam and/or hydrogen). *Notice that the temperature of the impactor was not modified for the 6 VERCORS tests, this parameter having been studied in earlier HEVA tests.*

Most of these tests have been preceded by an oxidizing plateau with mixed steam and hydrogen flow at a temperature around 1600 K in order to oxidize fully the cladding before the last heating ramp to the final high temperature plateau. The test matrix (table I) shows how the program has been implemented.

VERCORS 1 and VERCORS 2 were performed in mixed steam and hydrogen flow up to 2150 K :

- with low flow injection for VERCORS 1 in order to study the effect of high FP concentration during the aerosol transport phase,
- with higher flow injection for VERCORS 2 and with four intermediate plateaus, between 1070 and 1770 K, in order to quantify fission gas and volatile FP releases in conditions similar to a LOCA.

The next four tests were all performed up to higher temperature, around 2600 K, just below fuel collapse, except VERCORS 6, where the high fuel burn-up sample led to liquefaction at this temperature level. The conditions were:

- a mixed steam and hydrogen atmosphere for VERCORS 3,
- pure hydrogen during the last high temperature plateau for VERCORS 4, but after a cladding pre-oxidizing phase,
- pure steam during the last high temperature plateau for VERCORS 5, but after intermediate plateaus between 1070 and 1770 K, as for VERCORS 2, in order to confirm volatile FP release rates in LOCA conditions,
- a mixed steam and hydrogen atmosphere for VERCORS 6, like VERCORS 3, but with a 60 GWd/tU fuel sample.

The high temperature plateau was maintained 30 min for the last three tests and only 15 min for VERCORS 3 because of a blockage of the loop on the last stage of the impactor.

3.2. FP release results

Total FP release of each VERCORS test are summarized in table I.

Concerning VERCORS 1 and VERCORS 2, performed at 2150 K, approximately the same level as the earlier HEVA tests, similar FP kinetics were measured for fission gases, iodine and cesium ; the total released fraction reached 20 to 40% for these elements. For VERCORS 1, this fraction is a little higher, due to a prolonged plateau at the final high temperature (17 min instead of 13 min) and the use of fuel with a slightly higher burn-up (43 GWd/tU instead of 38 GWd/tU). These elements aside, molybdenum, antimony, tellurium (and some trace of barium) had measurable release fractions, which are higher for VERCORS 2, due to more oxidizing conditions for molybdenum and to intermediate plateaus, allowing full oxidation of the cladding, for antimony and tellurium.

Due to the more severe conditions, the four complementary VERCORS 3 to 6 tests have led to useful extension of the FP data base. According to their releases at 2600 K, FPs could be classified into four categories [5] :

- **Usual volatile FPs, iodine and cesium** and, **in addition, antimony and tellurium**, with nearly complete release at this temperature level. A delay for the release of tellurium and antimony have been noticed and identified by trapping in the unoxidized cladding (measured by gamma emission tomography for tellurium), but the release of these two elements reaches rapidly the level of iodine and cesium and seems even eventually to overtake them.
- **Semi-volatile FPs**, composed of **molybdenum, rhodium and barium**, with significant release, about half of the volatile FP release, but with low volatility chemical forms deposited close to the fuel, and with high sensitivity to the oxidizing or reducing conditions ; for instance the release of molybdenum is increased in oxidizing conditions due to the formation of volatile oxides MoO_3 (92% released in VERCORS 5, instead of 47% in VERCORS 4). On the other hand the release of barium and rhodium is increased in reducing conditions (respectively 45% and 80% of rhodium and barium released in VERCORS 4, as opposed to 20% and 55% in VERCORS 5).
- **Low volatility elements**, composed of **ruthenium, cerium, neptunium** and probably **strontium** and **europium**, with a low but measurable release, between 3% to 10%, exclusively deposited in the high temperature section of the loop, very close to the fuel. Reducing conditions (VERCORS 4) seem to increase the release of neptunium and cerium. No effect of the atmosphere was noticed for ruthenium in these conditions, an element which is known to have a high release, similar to volatile FP release, in high oxidizing conditions, for instance in air ([6], [7]).
- **Non-volatile FPs**, composed of **zirconium, niobium, lanthanum** and **neodymium**, with no measurable release in this temperature range under these conditions. Among these, in more severe conditions up to fuel melting temperature as recently performed in VERCORS HT and RT tests, niobium and lanthanum have been quantified as having a significant release [8].

Other elements, like uranium, not measurable by gamma spectrometry, were detected on impactor plates by SEM/EDS, but could not be quantified precisely in terms of released fraction.

4. VERCORS 6 results

The VERCORS 6 test, the last of the series, was performed with high burn-up fuel of 60 GWd/tU ; this section gives some results of this test concerning fuel degradation and FP release.

4.1. Fuel degradation

Compared to the three previous tests performed at the same temperature of 2600 K, VERCORS 6 was the first test leading to early fuel collapse during the high temperature plateau. Figure 3 shows, versus time, the fuel temperature and the total gamma activity signal, measured by the on-line spectrometer on the fuel. After a continuous decrease during the heat up phase and the beginning of the high temperature plateau, corresponding to volatile FP release, a sudden and large signal increase is noticed after 20 minutes at 2600 K ; it corresponds to the collapse of the fuel and its bulk relocation at the bottom of the crucible (*the next decrease of the signal at 18:40 is the result of a voluntary collimator change in order to reduce the count rate which saturated the electronic acquisition equipment*).

Figure 4 compares the X-ray radiographs performed respectively on the fuel samples of VERCORS 3, 4, 5 and 6 tests. In the three first tests, the fuel sample has maintained its integrity, even if many cracks could have been observed, especially on the VERCORS 4 sample. On the other hand, the VERCORS 6 fuel sample is severely damaged. The upper part of the radiograph is composed of the upper unirradiated half-pellet, which seems to have maintain its original dimensions, and two irradiated pellets (the upper and central ones), which have become thinner and have partially melted. The lower part of the radiograph is composed of the crucible retaining the remaining part of the fuel sample : the lower unirradiated half-pellet, which seems also to have maintained its integrity, and the lower irradiated pellet, which is totally melted and has interacted with the crucible.

Ceramographic examinations of these samples and complementary SEM/EDS analysis have confirmed the significant melting of the original irradiated fuel pellets and the strong interaction of the corium with the crucible, the zirconia sleeves surrounding the crucible, and even the tungsten susceptor.

4.2. FP behavior

The volatile FP release, almost complete for the four tests performed at 2600 K, is not particularly affected by the effect of the burn-up, except possibly faster kinetics, an effect which will be soon confirmed. Concerning semi-volatile and low-volatile FPs, the more severe degradation of high burn-up fuel does not increase their release ; the liquid corium phase formation seems even to retain a fraction of them in comparison with solid fuel (respectively 79%, 29% and 0,6% of molybdenum, barium and ruthenium released in VERCORS 6, as opposed to 92%, 55% and 6% in VERCORS 5).

Figure 5 gives the distribution of the main semi and low volatile FPs, compared to non-volatiles, inside the two samples recovered after the test and along the upper part of the zirconia sleeves, where the deposit of these semi and low-volatile FPs is significant (22% and 11% of the initial inventory for molybdenum and barium, roughly one third of the total released fraction).

Gamma emission tomography was performed on the sample containing the crucible and the remains of the melted lower pellet. Figures 6 compares ^{95}Zr and ^{103}Ru distribution, where it can be noticed that ^{103}Ru, possibly associated with tungsten within a metallic phase, is not located within the fuel and oxide phase, represented by ^{95}Zr ; it appears to surround the fuel, confirming its specific behavior already evidenced by gamma emission tomography of the PHEBUS FPT1 bundle [9].

5. VERCORS HT-RT program

Since 1996 a new VERCORS HT and RT program has been launched to improve the data base of fission product and actinide releases during the later phases of an accident, in particular up to fuel melting [10].

Compared to the VERCORS facility, the new hot cell apparatus VERCORS HT (HT for High Temperature) is rather different regarding the furnace, made of thoria, and the instrumentation (figure 7).

In particular, on-line instrumentation has been improved with :

- A thermal gradient tube (TGT) 0.7 m long, just downstream of the fuel furnace, devoted to the study of vapor-phase and aerosol deposition ; the axial temperature profile decreases from 1300 K to 300 K,
- The location of the impactor on a branch circuit in order to operate in a more suitable mode during a predefined period of the experiment instead of the full duration. When the impactor is not open, the aerosols are collected in a high capacity filter composed of granular beds.
- a specific iodine filter, separating the chemical species, in particular its molecular form,
- four gamma spectrometers, instead of three, to give information on five locations : the fuel, the TGT, the impactor, the filter and the gas capacity together.

The RT version of the loop (RT for Release of Transuranics) is more compact. The furnace is similar to the HT furnace, but its handling is easier ; thus it enables the frequency of the tests to be increased. In this simplified configuration, all FP and transuranic elements are trapped as near as possible from their emission point in a total filter. The release quantification of actinides and pure b FPs is then obtained by post-test chemical analyses, such as ICP-MS, alpha spectrometry, etc.

Table II gives the test matrix parameters as defined today.

- HT1 was performed in June 1996 in a reducing atmosphere. The fuel was heated up to 2900 K and collapsed early during the last heating phase at around 2600 K. The final report of this test has been recently released.
- RT1 and RT2, performed in 1998 without re-irradiation of the fuel, were aimed in part at measuring, with high precision the temperature of fuel collapse ; RT1 was performed with UO2 fuel while RT2 was performed with MOX fuel [10].
- RT5 was performed in 1998 with re-irradiated high burn-up fuel in order to confirm aspects of the VERCORS 6 test, notably the early fuel collapse.
- RT4 was performed and RT3 will soon be performed with a configuration of a debris bed to study the late phase of an accident, notably fuel volatilization, complementary to the PHEBUS FPT4 test. RT4 used unre-irradiated fuel (UO_2 and ZrO_2 fragments) and RT3 will use re-irradiated fuel fragments (only UO_2 to achieve higher temperature and quantify, in particular, the fuel volatilization rate).
- RT7 will be the second test with MOX fuel and the first using a re-irradiated MOX sample. Compared to RT2, it will be performed in reducing conditions, instead of the mixed steam/hydrogen of RT2.
- Finally, HT3 will be the second HT test. Compared with HT1, it will be performed under reducing conditions and with additional injection of boric acid and SIC (silver, indium and cadmium) within the fluid flow.

6. Conclusion

The VERCORS program represents a significant step forward in knowledge and accuracy of in-vessel source term data. Following the HEVA program, which contributed mainly towards data on volatile FPs, the VERCORS tests have extended the data base up to 2600 K :

- confirming the nearly total release of volatile species Cs, I, Te and Sb
- measuring the release of low volatile species and classifying them in three categories : "semi-volatile" (Mo, Rh, Ba), "low-volatile" (Ru, Ce, Np, Sr, Eu) and "non-volatile" (Zr, Nb, La, Nd).

The VERCORS 6 test, performed with high burn-up fuel, leads to early fuel collapse and partial liquid corium formation. Nevertheless, FP release did not increase significantly compared to solid fuel ; the liquid phase seems even to retain a fraction of some semi and low-volatile FPs.

The on-going experiments VERCORS HT and RT focus on the release of low volatility FPs and actinides, and will confirm or extend somewhat the data base at higher temperatures including the consideration of a wider range of parameters such as the nature of the fuel (UO_2 and MOX), its morphology (intact pellets or debris fragments), its burn-up, the impact of control materials (SIC and boric acid) and the conditions of the accidental sequence (oxidizing or reducing).

References

[1] : J.P. LEVEQUE, B. ANDRE, G. DUCROS, G. LE MAROIS, G. LHIAUBET, "*The HEVA experimental program*", Nuclear Technology, Vol. 108, Oct. 1994.

[2] : G. DUCROS, B. ANDRE, M. TOURASSE, D. MARO, "*The fission products and actinide release at high temperature in PWR fuel rod : the VERCORS safety program*", IAEA TC meeting on behaviour of LWR core materials under accident conditions, Dimitrovgrad, 1995

[3] : M. TOURASSE, B. ANDRE, G. DUCROS, D. MARO, "*Overview of the VERCORS program devoted to safety studies on irradiated PWR fuel*", Material Chemistry'96, Tsukuba, 1996

[4] : G. DUCROS, E. CONFORT, P. DREVON, "*Non destructive location of fission products in a damaged fuel rod section using gamma-emission tomography*", IAEA TC meeting on recent developments on post-irradiation examination techniques for water reactor fuel, Cadarache 1994

[5] : G. DUCROS, B. ANDRE, M. TOURASSE, D. MARO, "*The VERCORS safety program, source term analytical study with special emphasis on the release of non volatile fission products and transuranic elements*", CSARP meeting, Bethesda, 1995

[6] : F.C. Iglesias et al., "*Ruthenium release kinetics from uranium oxides*", ICHMT seminar on fission product transport processes in reactor accidents, Dubrovnik, 1989

[7] : D.S. Cox et al., "*Fission-product releases from UO2 in air and inert conditions at 1700-2350 K : analysis of the MCE-1 experiment*", A.N.S. I.T.M. on the safety of thermal reactors, Portland, 1991

[8] : G. DUCROS, M.P. FERROUD-PLATTET, M. BAICHI, P.P. MALGOUYRES, C. POLETIKO, H. MANENC, D. BOULAUD, M. DURIN, "*Fission product release on VERCORS HT1 experiment*", CSARP meeting, Albuquerque, 1999

[9] : B. CORNU, I. GIACALONE, "*Advanced non destructive examinations on severely degraded fuel from PHEBUS tests*", CSARP meeting, Albuquerque, 1999

[10] : P.P. MALGOUYRES, G. DUCROS, M.P. FERROUD-PLATTET, M. PROUVE, D. BOULAUD, "*The VERCORS HT facility for studies up to molten fuel conditions*", European working group "hot laboratories and remote handling" meeting, Windscale, 1998

[11] : P.P. MALGOUYRES, M.P. FERROUD-PLATTET, G. DUCROS, C. POLETIKO, M. TOURASSE, D. BOULAUD, "*Influence of MOX fuel in fission product release up to meltdown conditions*", NURETH-9 ANS meeting, San Francisco, 1999

Test	VERCORS 1	VERCORS 2[a]	VERCORS 3	VERCORS 4[b]	VERCORS 5[c]	VERCORS 6
Date of test	11-1989	06-1990	04-1992	06-1993	11-1993	06-1994
Fuel						
PWR irradiation	Fessenheim	Bugey	Bugey	Bugey	Bugey	Gravelines
Fuel burn-up (GWd/tU)	42,9	38,3	38,3	38,3	38,3	60
Re-irradiation	Siloe	Siloe	Siloe	Siloe	Siloe	Siloe
Test conditions						
Max fuel temperature (K)	2130	2150	2570	2570	2570	2620
Atmosphere (end of test)	Mixed H2O+H2	Mixed H2O+H2	Mixed H2O+H2	Hydrogen	Steam	Mixed H2O+H2
Last plateau duration (min)	17	13	15	30	30	30
Steam flow rate (g/min)	0,15	1,5	1,5	1,5 - 0	1,5	1,5
Hydrogen flowrate (g/min)	0,003	0,027	0,03	0,012	0	0,03
FP released fraction (%)						
Sr				< 6	< 6	< 6
Y			17			
Zr				< 3	< 4	< 4
Nb						0,3
Mo		15	42	47	92	79
Ru			0,36	7	6	0,6
Rh			0,52	45	20	
Sb	2	7	69	97	98	96
Te	4	18	76	100	> 98	97
I	30	23	70	87	93	97
Xe	33	23	77	86	87	~ 100
Cs	42	30	70	93	93	97
Ba	4	4	13	80	55	29
La			< 4	< 3	< 3	< 3
Ce				3	< 3	0,2
Eu			< 6	< 5	< 3	< 4
Np	0,006	0,016	0,4	6	< 4	0,3
U[d]				~ 2	~ 2	
Pu[d]				~ 0,2	~ 0,2	

[a] Test with intermediate temperature plateaus at 1070, 1170, 1470, 1770 K for 32, 12, 37 and 30 minutes
[b] Test under hydrogen, but with oxidizing intermediate plateau under mixed H2O+H2 at 1670 K for 60 minutes
[c] Test under pure steam, but with intermediate temperature plateaus at 1070, 1270, 1570 K for 30, 30 and 70 minutes
[d] Approximate value from ICPOES measurement of aerosols recovered on impactor plates, correlated with 137Cs measurement

Table I : VERCORS test matrix and total FP released fraction

VERCORS tests	HT 1	RT 1	RT 2	RT 5	RT 4	RT 3	RT 7	HT 3
Date of test	June 1996	March 1998	April 1998	December 1998	June 1999	November 1999	Beginning 2000	End 2000
Fuel	UO2	UO2	MOX	UO2	UO2/ZrO2 debris bed	UO2 debris bed	MOX	UO2
Burnup (GWd/tU)	47	47	41	60	3 cycles	3 cycles	3 cycles	4 cycles
Re-irradiation	SILOE	No	No	OSIRIS	No	OSIRIS	OSIRIS	OSIRIS
Max fuel temperature (K)	2900	2570	2440	Fuel collapse	Fuel collapse	Fuel melting	Fuel melting	Fuel melting
H_2 (mg/s)	0,2	0,45	0,45	0,45	0,4	1,25	Reducing	Reducing
H_2O (mg/s)	0	25	25	25	14,6	1,25	conditions	conditions
He (mg/s)	8	0	0	0	0	0		
Main objective	H_2 atm., high temperature	RT reference test	MOX fuel	High Burnup	Phebus FPT4 support	Fuel volatilization	MOX fuel	SIC injection

Table II : VERCORS HT-RT test matrix parameters

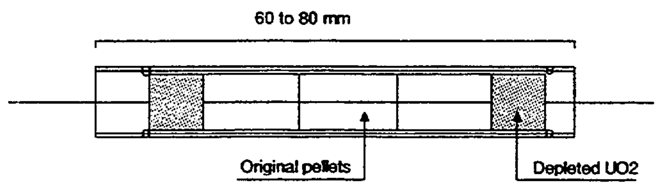

Figure 1 : VERCORS fuel sample

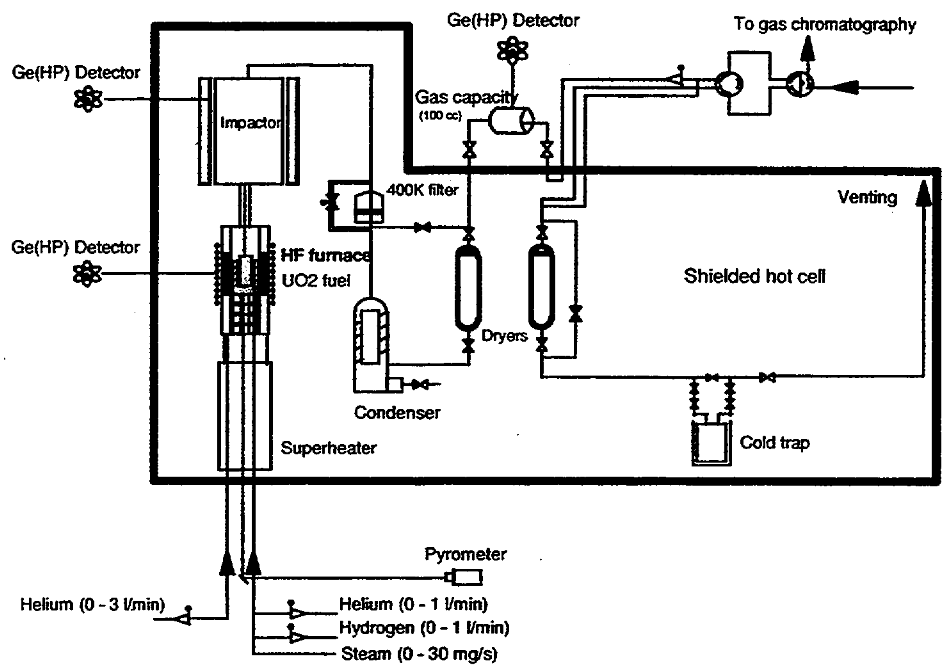

Figure 2 : VERCORS apparatus in hot cell

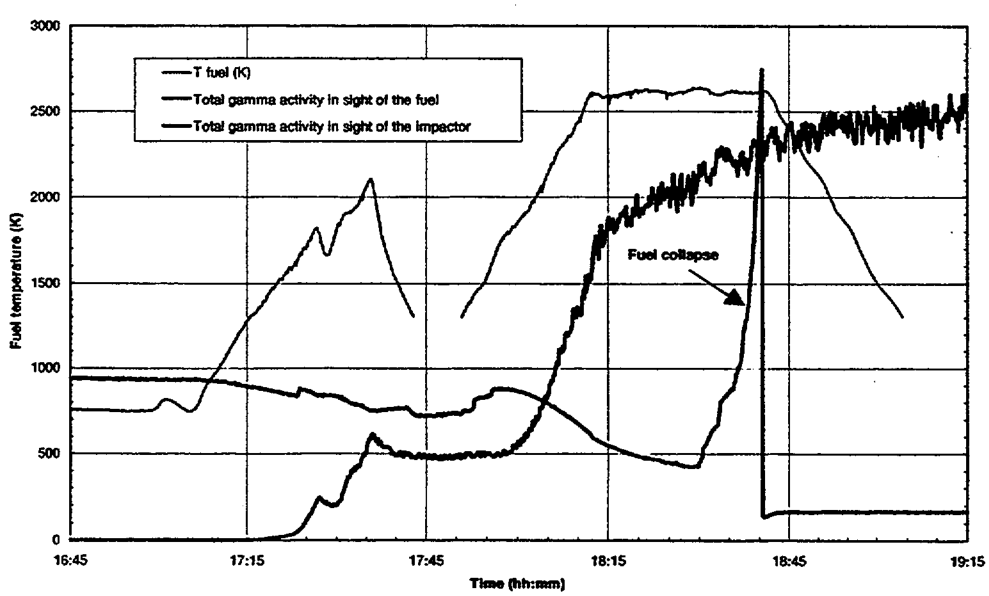

Figure 3 : VERCORS 6 fuel collapse

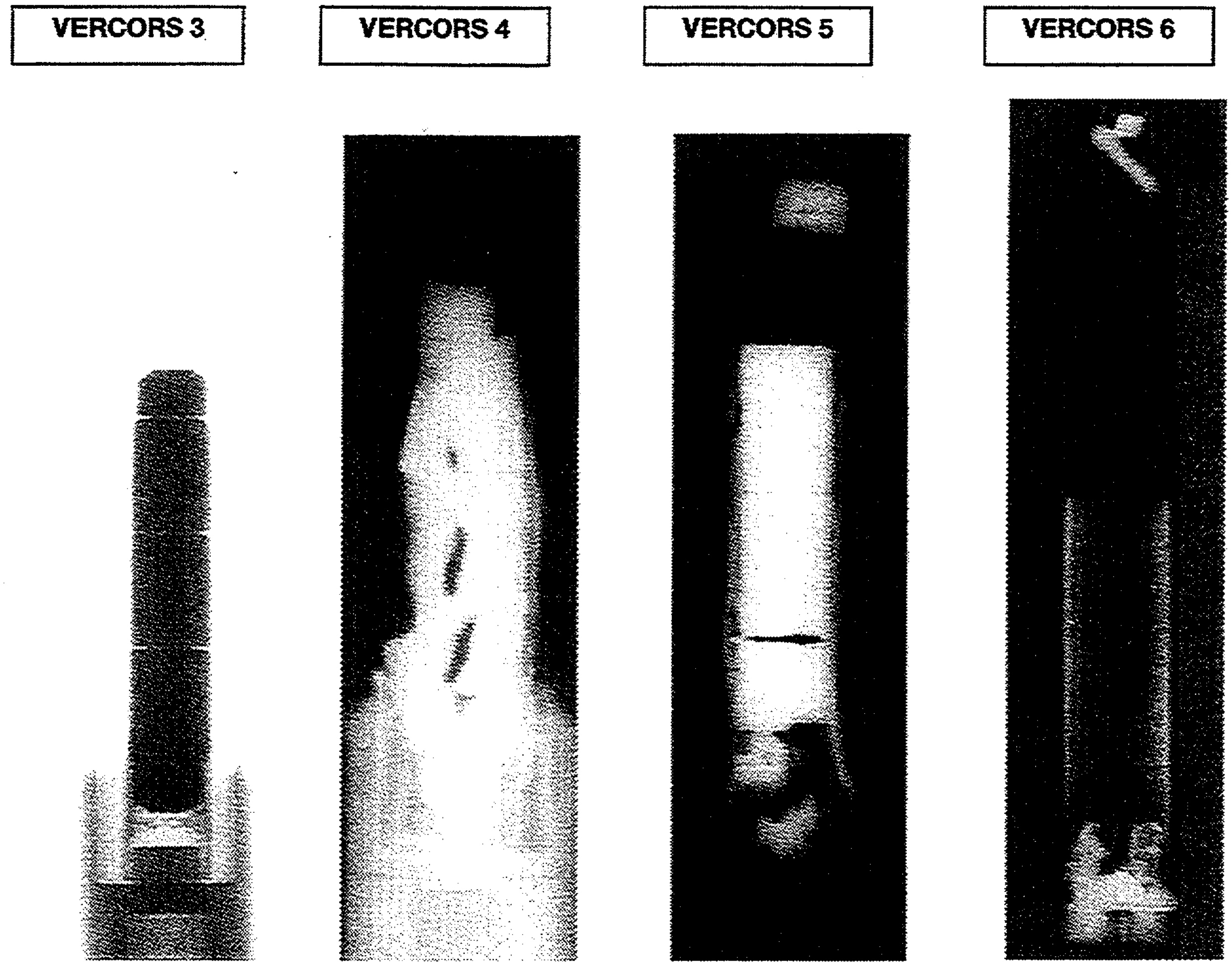

Figure 4 : X rays radiographs of VERCORS fuel samples after the tests

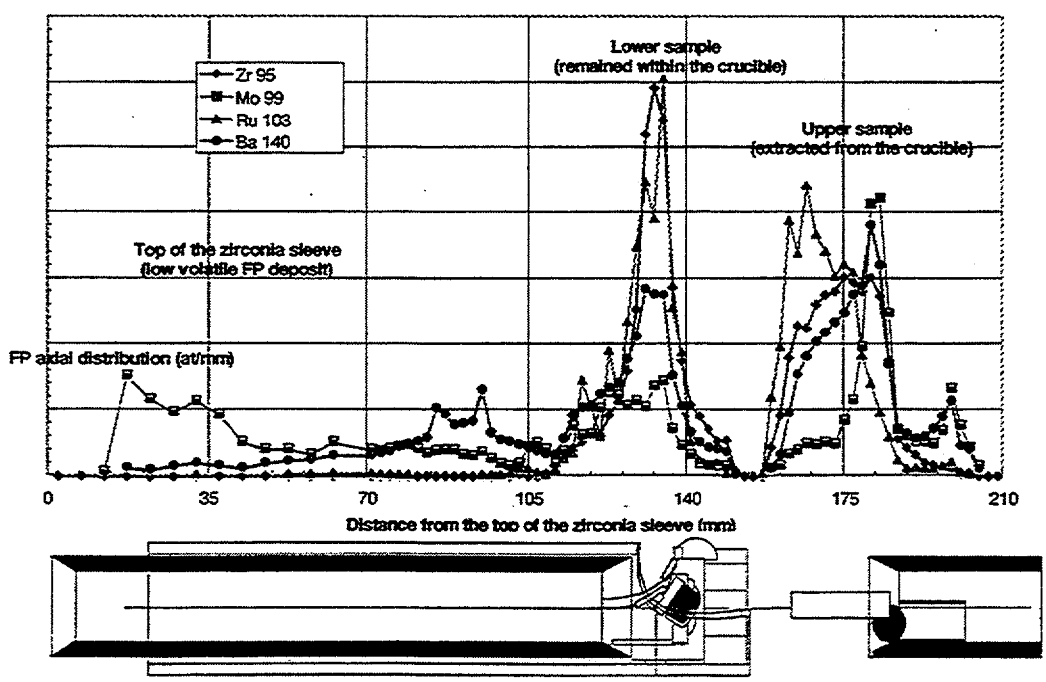

Figure 5 : FP axial distribution after VERCORS 6 test (crucible and zirconia sleeve)

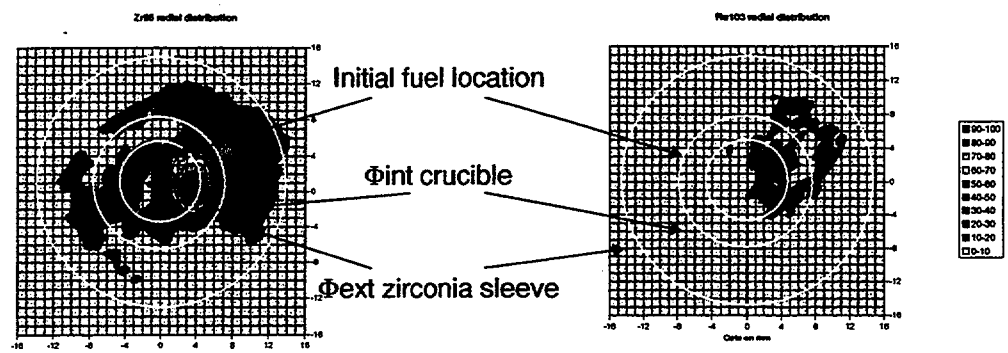

Figure 6 : ^{95}Zr and ^{103}Ru gamma emission tomography after VERCORS 6 test

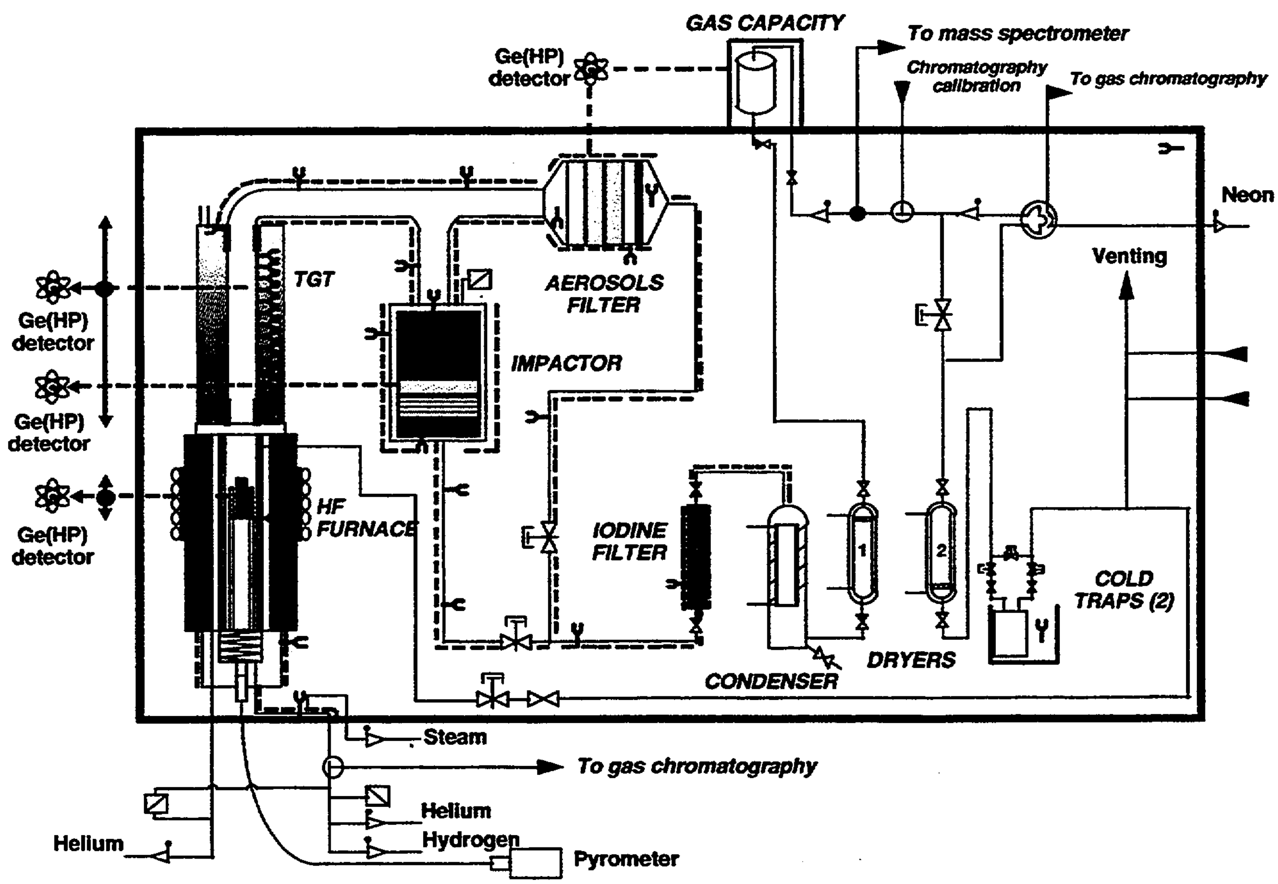

Figure 7 : VERCORS HT apparatus in hot cell

AeRosol Trapping In STeam generator (ARTIST): an Investigation of Aerosol and Iodine Behaviour in the Secondary Side of a Steam Generator

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Containment bypass sequences such as an accident involving a steam generator tube rupture (SGTR) with stuck-open relief valve represent a significant public risk by virtue of the open path for release of radioactivity. An ameliorating factor is deposition of fission products on the steam generator (SG) tubes and other structures in the SG secondary. The absence of empirical data, the complexity of the geometry and controlling processes, however, make the retention difficult to quantify and credit for it is typically not taken in risk assessments. The ARTIST experimental programme to be conducted at Paul Scherrer Institut, Switzerland, will simulate the flow and retention of aerosol-borne fission products in the SG secondary, and thus provide a unique database to support safety assessments and analytical models. The facility represents a Framatome designed SG, scaled 1:24 in area and number of tubes. Scaling of the break flow presents a particular challenge since the aerosol retention process operate at contrasting length scales - locally near the break and within the SG as a whole. Preliminary calculations have identified a baseline set of conditions and demonstrated the feasibility of the rig design and scaling principles. Flexibility of the rig and mode of operation enables simulation of a range of SG designs, accident situations (break configuration, aerosol characteristics, ...) and accident management measures such as SG refilling.

1. Background and Motivation

Steam generator (SG) tubing is subject to a variety of degradation processes that can lead to cracks, thinning and, potentially, rupture. Despite improvements in SG design, manufacturing and modes of operation, SG tube rupture (SGTR) events occasionally occur during PWR operation worldwide which underline the need to pay particular attention to SGTR sequences.

A particular safety challenge arises from an SGTR in combination with other failures such that a core melt occurs, in which case there may be a direct path by which radioactive fission products can be transported to the environment. Sequences of this kind are referred to as containment bypass and, despite their low probability, represent a significant or even dominant contribution to the overall public risk. Although probabilistic safety assessments (PSA) typically take little or no account of any retention of fission products in the secondary side, the complex geometry of the tube bank, support plates, separators and dryers provides a large surface area on which fission products may be trapped. The presence of liquid water in the SG bundle may further augment the retention. However, the processes that control the retention are complex and there are no reliable models or empirical data with which to perform assessments. The experiments to be performed by the Paul Scherrer Institut (PSI) in the ARTIST programme will be the first of their kind to provide a database for retention under the range of conditions that may apply in a reactor accident. Reduced uncertainty in predicted radionuclide release will enable plants to reassess their PSA level 2 results and consequently the assumptions used in level 3.

The ARTIST programme is motivated by the concurrence of interest between KKB, following a PSA study which identified the SGTR with consequent stuck-open relief valve as the risk-dominant sequence, and PSI whose role is to address current concerns and to support ongoing improvements in nuclear safety. Of particular interest in this latter category is investigation the effectiveness of refilling the faulted SG in order to arrest the release of radionuclides.

2. Reference plant accident sequence

Release and transport of fission products from a degraded core, through the primary to the secondary side and, potentially, to the environment may occur as a consequence of a variety of sequences, typically falling into one of three groups:

- a design basis SGTR with other failures leading to core damage;
- a core damage sequence which causes the SG tubes to be subjected to large thermal and pressure loads;
- a design basis SGTR in which mildly contaminated primary coolant (iodine-spiked) is carried over in conditions outside the normal operating range of the SG.

In this section we concentrate on a sequence within the first group, previously analysed in a level 2 PSA study for KKB [1] and identified as a major risk contributor. The case is here referred to as the reference transient, the results of which are used to define a baseline set of conditions for the ARTIST experiments.

The assumed initiating event is a double-ended guillotine break near the bottom of the hot side of one of the SG tubes. (KKB is a two-loop PWR; for convenience the faulted and intact SGs are referred to as SGA and SGB, respectively.) The reactor protection, emergency coolant systems (ECS) and auxiliary feedwater (AFW) systems function normally. However, it is assumed that no operator-initiated measures (such as spray operation to equalise the primary and secondary pressures) are taken in the early stages of the sequence, and that overfilling of the faulted SG secondary causes the relief valve (RV) to stick open, hence leading to a long term loss of cooling and a consequent core melt and release of fission products.

The thermal-hydraulic and core degradation was analysed using SCDAP/RELAP5 [2,3]. The results for the fuel temperatures were then used as input to the SASPROG, which incorporates the release correlations from NUREG-0772 [4], to calculate release of fission products from the core. The calculated event sequence is given in Table 1, together with the corresponding result from the PSA study, using MAAP. Despite differences between the modelling approach and the way in which some of the boundary conditions are defined, there is sufficient similarity between the calculations to provide confidence that the results give a credible description of the sequence and a meaningful basis for defining the experimental conditions. Among the essential features of the transient are:

- both the primary and secondary sides of the faulted SG are completely void before the core starts to heat up and remain so throughout the rest of the sequence;

- the intact SG is depressurised to one bar by operator action before the onset of core degradation;
- accumulation of hydrogen in the intact steam generator impedes the condensation of steam and limits the primary side depressurisation such that the flow is choked at the break.

Event	MAAP [1]	SCDAP	Notes on SCDAP	(key event
Tube rupture in SGA	0	0		
AFW initiation, scram, SGA isolation	91	100	SI signal at $p < 11.72$ MPa	
Power decay	91	102	SI + 2s	
SGA relief valve open	98	116	$p > 6.50$ MPa	
HPIS initiated	100	130	SI + 30 s	
Pressuriser empty	109	112	level < 2 m	
SGA relief valve closed	120	206	$p < 5.88$ MPa	
SGA relief valve cycling	121	350		
SGA AFW isolation	600	600	break + 600 s	
SGA relief valve sticks open	1052	1918	SGA full (level > 16 m)	(a
ECS tanks empty	19238	32377		
Accumulator initiation		32844	$p < 5.27$ MPa	
Accumulators empty	20830	37424		(b
Main coolant pumps tripped	30004	37703	upstream void > 10%	(c
SGA dried out	59129	60715	level < 0.1	(d
Core uncovery starts	66212	74318	max. temp. > 700 K	(e
SGB relief valve latched open	68945	76318	core uncovery + 2000 s	(f
Main coolant pumps restarted	72185	80318	core uncovery + 6000 s	
Molten pool forms	72187	83822		(g
End calulation		90533	48 % of core molten	

Table 1: Event sequence for reference SGTR transient

The transient conditions in the three coolant regions (reactor vessel, faulted and intact SG secondary sides) are illustrated in Figures 1 and 2, with the key events shown in Table 1 (a,b, ...) also indicated. Representative conditions during the period of interest are presented in Table 2.

3. Facility characteristics and capabilities

The test section will be connected to the PSI aerosol generation system (DRAGON) which provides a flow of single or multi-component aerosol in a steam or steam/non-condensable carrier gas. The aerosols are generated by feeding the materials in powder or liquid form at the desired rates to a plasma torch, in which they evaporate and then recondense in the stream of cooler carrier gas. The aerosol flow and characteristics can be controlled in a stable manner.

The ARTIST test section comprises several modules to simulate the SG secondary, as shown in 3. These comprise:

- a scaled (1:24) tube bundle of diameter 57.3 cm containing 264 straight tubes of outer diameter 19 mm and maximum height 3.8 m;
- an upper cap to simulate the U-bend segment of the SG tubes;
- a tube sheet plate
- three support plates spaced 1.1 m apart
- one separator unit of actual size
- one dryer cell of actual size

Parameter	Value
Time period of interest	80000 - 90000 s
Decay heat level	7 MW (0.5 MW)
Primary side total pressure	0.5 Mpa
Primary side steam pressure	0.1 - 0.4 Mpa
Secondary side pressure	0.1 Mpa
Break flow (hot side)	0.20 kg/s
Break flow (cold side)	0.05
Break fluid temperature	1000 K
SGA structure temperature	650 K
Fission product release rate	12 g/s

Table 2: Conditions at time of release

The bundle diameter is considered large enough to reproduce the jet behaviour as the momentum is dissipated after it emerges from the break, provided the break is not too close to the shroud and oriented towards it. The spacing between the support plates, the tube diameter, pitch and wall thickness are the same as in the real SG. The tube length is just 40% of the real unit due to building limitations, so there are just three bundle stages instead of nine as in the real plant. This is believed not to have a major impact as the flow is expected to become fairly evenly distributed beyond the break stage, so that the decontamination factor (DF) will be nearly the same in each of the subsequent stages. Results from measurements of the "far-field" DF are expected to extrapolate to more stages. However, the reduced height implies a shorter residence time which might be important for other retention processes, such as thermophoresis. This can be addressed by reducing the carrier gas flow rate. Finally, it is noted that the bank of active tubes is not represented. A single full length U-tube is envisioned for separate effect experiments on aerosol retention, principally by turbulent deposition, inside the tube, to provide data on the fraction of the aerosols entering the SG that reach the break. The support plate, tube layout, possible locations of the broken tubes and the full length U-tube are shown in Figure 4. The separator and dryer units are full scale and identical to those in the real SG. Therefore the results for retention on will be directly applicable as long as the aerosol and thermal-hydraulic conditions are reproduced correctly. The heights of these components above the top of the bundle and the flow area of the pipe connecting the shroud and separator inlet are also preserved.

Parameter	Beznau	ARTIST
Number of tubes	3238	264 a
Number of dryers	12	1
Number of separators	12	1
Maximum height of tube (m)	9.0	3.8 (9.0 b)
Bundle diameter (m)	2.68	0.57
Bundle total area (m^2)	5.641	0.258
Bundle flow area (m^2)	3.790	0.185
Support plate flow area (m^2)	1.288	0.052
Support plate flow area per tube (cm^2)	1.97	1.97
Bundle : support plate flow area	2.94	3.56
Surface : volume (m^{-1})	102.2	87.2
Bundle hydraulic diameter (cm)	3.1	3.1
Bundle porosity	0.67	0.72

a: straight (equivalent half U-tube b: tube length for cold side break

Table 3: Beznau : ARTIST scale parameters

It is understood that, although the rig configuration is based on a specific reference plant, all vertical U-tube SGs that are deployed in commercial PWRs are qualitatively and conceptually similar, despite differences in parameter values (tube diameter, etc.). Simulation for different plant designs will require adjustment of the experimental boundary conditions to take account of the relevant plant parameters as well as the accident definition, in order to best preserve the thermal-hydraulic and aerosol dimensionless groups. The DRAGON and ARTIST facilities can be operated over a range of conditions of pressure up to 4.5 bars, a carrier gas total flow rate from 60 to 600 kg/h (with a possibility to increase to 850 kg/h), steam flow rate from 50 to 300 kg/h, and temperature up to 600 °C. The non-condensable gas may be Air, N_2, Ar or He. Aerosol loading up to 10 g/m^3, a mix of 4 components which may be soluble and insoluble, liquid and solid, and aerodynamic mass mean diameter (AMMD) from 0.5 to 5 µm can be achieved with the DRAGON facility. The scaling with respect to KKB is summarised in Table 3. In the discussion that follows, the reference transient is used to define a baseline or nominal set of conditions for the experiments.

4. Major phenomena in SG secondary

The controlling phenomena depend on whether the reference sequence is a severe accident in which case the secondary side conditions may be dry or wet, or a design basis accident in which

case the primary coolant is single phase liquid but superheated with respect to the secondary side.

4.1 Dry SG in severe accident conditions

The following mechanisms account for aerosol retention in dry conditions:

- turbulent deposition inside the ruptured tube;
- inertial and turbulent deposition in the secondary side;
- gravitational settling;
- agglomeration;
- thermophoresis

4.1.1 Turbulent deposition inside the ruptured tube

The reference calculation suggests that choked flow conditions at the break are typical, with velocities in the broken tube of a few 100 m/s. Previous studies [5] indicate that turbulent deposition is the dominant mechanism at such velocities and the retention is primarily a function of the dimensions (L/D) for particles of interest (> 0.3 µm). Retention may be offset to some extent by resuspension, to an extent which is as yet uncertain. Therefore it is important to preserve the flow and aerosol characteristics so that the results are directly applicable.

4.1.2 Impaction and turbulent deposition in the bundle

While these mechanisms are not the same, they exhibit a similar dependence on the flow velocity and so they can be lumped together for the purposes of scaling. Inertial deposition is due to the inability of particles to follow the flow around obstacles, while turbulent deposition occurs on surfaces that are parallel to the flow as a result of the large eddy velocities in highly turbulent flow.

Inertial and turbulent deposition in the break stage

For choked flow at the break, the local flow velocities are expected to be of the order of 100 m/s or greater and, for a horizontal break predominantly across the nearby tubes. The aerosol retention deposition on a single cylinder has been correlated by Douglas [6] as a function of the Stokes number, Stk, defined as

$$Stk = \rho_p d_p U C_S / (18\mu D) \equiv \tau U/D$$

where ρ_p and d_p are the particle destiny and geometric diameter, U the gas velocity, C_S the slip factor, μ the viscosity, D the cylinder diameter, and τ is the particle relaxation time. The collection efficiency varies between 2% and 30% as Stk varies from 0.004 to 0.06. Computational fluid dynamics (CFD) analyses show that the full nominal flow the velocity decreases from 300 m/s at the break to 10 m/s over a distance spanned by a few tube rows [7]. Table 4 below gives the expected Stokes numbers for these conditions.

AMMD (μm)	τ (s)	Stk: U=300 m/s	Stk: U=10 m/s
1	5.8e-6	0.09	0.003
3	4.6e-5	0.72	0.024
10	3.8e-4	6.0	0.200

Table 4: Stokes numbers for break stage

From these results it is clear that a significant fraction of the particles will be retained on the nearby tubes, but the retention decreases considerably away from the break.

It is noted that, in order to reproduce the aerosol retention locally, the flow rate should be the same as the full break flow. This results in an average bundle axial velocity of 4.1 m/s. However the average velocity in the bundle under the nominal conditions would be simulated better by scaling the flow rate according to the rig/plant flow area (1:20.5) to give a velocity of 0.2m/s. These two contrasting scale ratios have implications for simulating the flow and aerosol retention in the main part of the bundle. In particular the measured retention in the break stage will include a contribution due to impaction at the support plate.

Impaction on the support plates

Away from the break the velocity decreases by a large factor as the flow spreads out and moves upwards towards the support plates. The flow is then channelled through the small openings that surround the tubes where inertial impaction occurs at the orifices. Again, the fraction retained is a function of the local

Stokes number (based on the orifice effective diameter, 0.79 cm, as length scale), which are evaluated for the scaled flow rate in Table 5.

AMMD (μm)	$Stk^{1/2}$: $U_{,avg}$=0.2 m/s	$Stk^{1/2}$: $U_{,avg}$=4.1 m/s
1	0.012	0.055
3	0.033	0.155
10	0.098	0.444

Table 5: $Stk^{1/2}$ values for flow at support plate

Experiments by Ye and Pui [8] show that there is very little retention for values of $Stk^{1/2} < 0.2$. Simple extrapolation suggests that retention at the support plate is negligible at the scaled flow rate, and is significant only for the largest particles at the full flow, but some caution should be used due to the complex geometry and the non-uniform flow conditions below the support plate. Nevertheless, impaction on the plate is expected to make only a small contribution to the total retention in the break stage, even at the full flow.

Impaction on the tubes in the far field stages and U-bends

The flow beyond the break stage is mainly in the vertical direction, and little retention by turbulent deposition is expected on the tubes with a mean velocity of 0.2 m/s. Impaction at the U-bends can be estimated from the cross-flow data of Douglas [6] which show no retention at values of $Stk^{1/2} < 0.06$. Table 6 shows the retention is negligible for the aerosol sizes of interest even

AMMD (μm)	$Stk^{1/2}$: U= 0.1m/s	$Stk^{1/2}$: U=0.2 m/s	$Stk^{1/2}$: U=0.4 m/s
1	0.004	0.008	0.016
3	0.011	0.022	0.044
10	0.032	0.063	0.126

Table 6: Stokes numbers for U-bends

for a wider range of velocities. There is some latitude, then, to choose smaller or larger flow rates in order to respectively match the total residence time in the bundle and the velocity in the separator and dryer regions.

Impaction in the separator and dryer sections

The ARTIST model will contain one full size separator and one dryer instead of the 12 (each) units in the KKB SG. To preserve the mean velocity and residence time for the nominal conditions the flow rate should be scaled by 1/12, that is 75 kg/s. This corresponds to a velocity of ≤0.33 m/s in the separator tube and a residence time of ≤9 s, which is too short for agglomeration or settling to be significant.

The flow exits the separator upper lid and enters the open header where the mean velocity decreases to a few cm/s, so that impaction underneath the dryer is negligible. The residence time is about 8 s in this region so that, again, agglomeration and settling are insignificant.

The free flow area per dryer panel in the KKB SG is 0.3982 m^2, compared with 0.3586 m^2 in the ARTIST facility. To preserve the flow rate, the velocity in the ARTIST channel should therefore be reduced by 11%. However, due to the small magnitude of the velocity the retention due to impaction in the channel is small. Therefore retention by impaction in the separator and dryer sections will be replicated by a 1/12 flow scaling. From the discussion concerning impaction in the bundle, the same scaling can be applied throughout the SG, except near the break where the full flow is appropriate.

4.1.3 Agglomeration

Agglomeration occurs and modifies the size distribution if the concentration and residence time are large enough. According to the KKB plant calculations the concentration of volatile fission products is about 10 g/m^3, which results in a number concentration of 2.7 x 10^7 /cm^3, assuming spherical particles of initial size 1 µm and density 2 g/cm^3. Assuming a total residence time of 60 s in the KKB SG secondary, the theory of monodisperse agglomeration [9] predicts the number concentration to decrease by about one-third, in the absence of other mechanisms. The particle diameter is then increased by 16%, a small but not negligible change. The effect will be greater for lower flow rates and/or smaller initial sized particles, while smaller for higher flow rates and/or larger particles.

4.1.4 Gravitational settling

The settling velocity varies from 0.2 to 18 cm/min over the AMMD size range of 1 to 10 µm. Gravity setting is negligible for a residence time of the order of one minute, and therefore uncertainty in the aerosol shape factor is not an issue.

4.1.5 Thermophoresis in the far field stages

The incoming gas may be some hundreds of degrees hotter than the bundle structures, in which case thermophoresis is an important removal mechanism. The thermophoretic velocity primarily depends on thermal gradient and is a mild function of particle size [8]. Table 7 shows the expected thermophoretic velocities, assuming a maximum temperature difference of 500 °C and a representative distance of 1 cm between the subchannel centreline and tube wall. Since the residence time in a single stage is about 6 s, assuming an average flow velocity of 0.2 m/s, the thermophoretic displacement is of the order of 0.3 cm which is comparable with the distance to the tube wall. Thermophoresis is therefore likely to be significant. Choice of a lower flow rate to reproduce more closely the residence time in the plant SG will result in yet more retention.

AMMD (µm)	$vel_{,thermo}$ (cm/s)
1	0.065
3	0.050
10	0.039

Table 7: Thermophoretic deposition velocities

4.1.6 Discussion of residence times in a dry SG

From the above discussion and the geometry of the ARTIST facility, it is possible to identify 5 areas of investigation with 5 corresponding time scales.

1. Inside the broken tube: highly turbulent conditions with velocities of up to 200 m/s occur in the broken tube, and can be directly reproduced using the full flow rate. Conditions upstream of the break will be preserved since a single full length U-tube will be included, while the break can be located at the hot side near the entrance, on the cold on near the exit, or anywhere in the tube span. The main retention mechanism is turbulent deposition, which depends mainly on velocity. Residence time is of no consequence.
2. Near the break: within the break stage the flow and hence velocity should be scaled 1:1. The main retention mechanisms are turbulent deposition and inertial impaction, which depend mainly on velocity and are essentially independent of residence time.
3. Bundle region away from the break: beyond the break stage the flow will redistribute itself more or less uniformly, with a typical velocity of 0.2 kg/s which corresponds to a nominal flow of 900 kg/h for the KKB SG. The dominant removal mechanism is expected to be thermophoresis with smaller contributions from impaction at the support plates and U-bends. Two scale ratios apply. 3.1 Thermophoresis: the flow should be scaled by 1:49 to preserve residence time. 3.2 Impaction: the flow should be scaled by 1:21 to preserve the mean velocity
4. Separator and dryer: ARTIST represents the 12 separator units and dryer panels with a single unit and panel, so the flow should be scaled by 1:12 to preserve the velocity and residence time.

The above scaling considerations are summarised in Table 8. It might not be necessary, however, to perform experiments at each of these flow scalings. It may be possible to lump the study of impaction in the bundle far field and separator/dryer (3.2 and 4) by using a scaling of 1:12. This would have the advantage of minimising thermophoresis. Impaction in the bundle far field is expected to be small so the scaling compromise is unlikely to result in significant distortion. Alternatively, a scaling of 1:21 might be used to lump mechanisms 3.1, 3.2 and 4 to seek an integral simulation of aerosol retention throughout the far-field of the secondary side.

Region	Main phenomena for investigation	Quantity to preserve	Typical gas flow rate (kg/h)	Remark
Inside broken tube	turbulent deposition	velocity in tube	full flow	
Bundle near-field (break stage)	turbulent deposition, impaction on tubes	velocity field near break	full flow	
Bundle far-field (away from break)	impaction on support plates, U-bends	mean velocity	1/20th of full flow	thermophorsis to be avoided
Bundle near- and far-field	thermophoretic deposition on tubes	residence time, ∇T	1/49th of full flow	minimise inertial effects
Separator-dryer	impaction on structures	mean velocity	1/12th of full flow	thermophorsis to be avoided

Table 8: Major aerosol phenomena and corresponding gas flow rates in dry SG

4.2 Wet SG in severe accident conditions

A possible AM measure is to refill the faulted SG in order to re-establish heat removal and to provide a pool where the incoming aerosols can be scrubbed. Data from the POSEIDON experiments [10] conducted at PSI indicate that a DF of 10 or more can be obtained even in a shallow pool. Several factors apply in a refilled SG that would suggest even more effective pool scrubbing.

There are three distinct regions in gas-pool interactions:

- the immediate injection zone characterised by formation of a gas jet or globule depending on the injection conditions;
- the break-up zone where the jet or globule disintegrates into smaller bubbles
- the bubble rise zone where the bubbles rise at locally terminal velocity and, in the present context, periodically squirt out through the holes in the support plate.

Beyond the break-up zone (typically 10 globule diameters from the injection site), the bubbles are oblivious of the details of the injection process; therefore the scrubbing in the bubble rise zone is independent of the carrier gas flow rate

In addition, some of the incoming steam condenses almost immediately, with scrubbing of a corresponding fraction of the aerosols. The extent of condensation depends on the temperature of the water pool. If the pool is close to boiling, which is likely due to the hot structures and continued steam injection, condensation is minimal. Therefore the main removal mechanisms are inertial, namely:

- jet impaction at the injection site
- centrifugal impaction and gravitational settling during bubble rise

The first mechanism depends on the injection flow rate and is replicated by adopting the full flow. It is supposed that the jet momentum is locally dissipated as in the real SG and is not affected by the presence of the shroud. This is supported by past experience [10] which indicates that the jet is generally confined near the injection site. The removal mechanisms during bubble rise are reproduced without further scaling as they are independent of injection conditions.

It is noted that the timing of refill may be important. The above discussion concentrates on refilling prior to significant release of aerosols from the core. The refilling process itself is likely to be characterised by rapid cooling of the hot structures, and hence violent boiling, high steam flow and droplet entrainment. Refilling of a hot SG after significant aerosol release has already taken place may result in some resuspension of previous deposited aerosols, and transport of aerosols with the entrained droplets. The opportunity exists to address the possible impact of deferred SG refilling on release.

4.3 Wet SG in design basis accident conditions

The focus of DBA conditions, which involves the flow of mildly radioactive primary coolant to the secondary side, is on the potential release of radioactive iodine to the environment. Assumptions are typically made [11] concerning:

- the level of contamination ("iodine spiking") of the primary coolant prior to the SGTR;

- the primary and secondary thermal-hydraulic conditions which determine hence the fraction of coolant which flashes at the break;
- the iodine partition coefficient in the pool.

Three mechanisms are identified for iodine release to the environment.

1. Iodine partitioning: as break flow of primary coolant continues, iodine in the pool partitions between the liquid and vapour phases at a fraction determined by the prevailing conditions.
2. Primary flashing: a fraction of the primary coolant flashes to steam and the associated iodine is transported with the steam. Some of the iodine will be scrubbed in the pool or removed at the upper structures. In the case of a break above the pool surface, iodine may also be transported with liquid droplets carried with the steam.
3. Droplet carryover: fine droplets are carried with the steam bubbles produced by flashing of the primary coolant, and released to the environment after some scrubbing in the pool and retention at the upper structures.

The effect of partitioning is difficult to reproduce in the ARTIST facility since the partition coefficient is sensitive to the pool thermodynamic state. However, analyses [11] have shown it to be of secondary importance.

Primary flashing, where iodine is transported in the vapour bubbles, can be studied in ARTIST under low pressure conditions. The results cannot be applied directly since the flashing process, the fraction and the bubble dynamics depend on the pressure and on other conditions. Extrapolation to a plant DBA may be possible via detailed analysis using appropriate computer models. Nevertheless, the data can be expected to yield useful bounding estimates provided the test conditions are defined suitably.

Droplet carryover has been shown to be a negligible source of release if the break is sufficiently below the pool surface [12]. However, a break can occur in the U-bend, in which case retention in the pool is minimal. Droplet retention in the upper sections is then important and can be examined in ARTIST. Special consideration must be given to the size distribution, since retention by the separator and dryer is mainly a function of carrier gas flow rate and droplet size.

5. Break configuration

The ARTIST facility provides for a variety of break configurations. Although the scaling effort has focussed on the double-ended guillotine break, evidence from actual SGTR events shows that fish-mouth breaks and narrow cracks are more common. These considerations will be kept in mind for the final test matrix, but at present many possible locations and geometry are accommodated in the design as shown in Figures 4 and 5. The flexibility of configuration will make it possible to address particular safety concerns and also investigate the influence of geometry on release. For example, break types 1 and 8 are at locations where breaks have previously tended to occur. An upward orientation is expected to reduce the crossflow and hence impaction on the nearby tubes; break types 3 provides the greatest potential for retention inside the tube; break types 4 and 5 are expected to bracket the effect of location on retention in the break stage. Other break configurations will address the effects of symmetry, multiple breaks which may be separate or interacting, etc. Regarding the last, however, the flow rate is limited to approximately that from a double-ended guillotine break.

6. Flow field in SG secondary

One of the main concerns as regards the scaling is to ensure that the shroud will not impact the aerosol retention as long as the break occurs away from the periphery or in the periphery but facing inwards. A CFD model was developed and used to determine the flow distribution [7] for three break configurations:

1. a fish-mouth horizontal break on a peripheral tube, oriented to the centre of the bundle;
2. an axisymmetric horizontal beak at the centre of the bundle;
3. an upwards oriented break at the tube sheet and in the centre of the bundle.

The results showed:

- uniform static pressure at the shroud, so that the wall has little impact on the jet behaviour;
- the region of high velocity is localised near the break and confined to a small volume;
- the flow at the support plate is distributed across the entire flow area, though not quite uniformly, with a maximum velocity of about 10 m/s at the plate (roughly 2.5 times the average).

Experiments by Douglas [6] show that aerosol impaction at the wall is important only for Stokes number >0.04, which corresponds to a velocity of >30 m/s for a 3 μm particle. Together with the CFD results, this shows that the retention in the break stage is not compromised by the presence of the wall.

Further analyses using CFD methods will be performed to support final specification of the test conditions and interpretation of the results.

7. Instrumentation

The main goal of the ARTIST experiments is to characterise aerosol and gaseous retention in each section of the model SG, by measuring the aerosol and iodine concentration and thermal-hydraulic conditions at the inlet and outlet of each section. Supporting data will be provided by sampling and use representative surfaces.

In addition to classical thermal-hydraulic instruments (pressure transducers, thermocouple, flowmeter, anemometer, etc.), different aerosol measurement systems are available and can be readily connected to the inlet and outlet lines of the various parts of the test section. The available devices are:

- 12-stage, low-pressure Berner impactors with cut-off diameters ranging from 0.009 μm to 16 μm.
- 8-stage, high-pressure, Mark III Andersen impactors with cut-off diameters ranging from 0.3 μm to 8 μm to provide redundant and confirmatory data.
- Calibrated photometers (Sigrist) to measure online aerosol concentration over a wide range, from 0.3 g/m^3 to 20 g/m^3.
- Semi on-line quartz crystal microbalance cascade impactors (PC-2 QCM, California Measurements), which measure online the aerosol size distribution and concentration. They provide redundancy for the online concentration measurement and also make it possible to monitor the changes in outlet aerosol size distribution and concentration during a transient such as the initiation of accident management measures.

- Millipore membrane filters to measure aerosol concentration over a wide range - 0.1 g/m^3 to 40 g/m^3
- Water and gas grab sample devices together with associated ion selective electrodes (ISE) for analysis of iodine and caesium content and acidity. The will enable more accurate mass balances for these important species.

Besides local measurements, an aerosol global mass balance will be performed. To facilitate the procedures, sections of the tube bundle will consist of removable units which can be analysed for aerosol deposits.

A preliminary instrumentation plan recognises three different requirement levels for data:

1. "Must have" devices necessary to obtain essential data from the tests.
2. "Should have" devices to provide more detailed data and enhance the value and applicability of the results.
3. "Complementary" devices for gathering additional but not necessarily crucial information from the tests, and which will depend on the resources of the programme.

Details of the preliminary instrumentation plan are shown schematically in Figures 6 and 7.

The proposed instrumentation will provide the following:

- The thermal-hydraulic conditions throughout the facility: carrier gas flow rate and composition, gas and wall temperature profiles, and pressure.
- Qualitative information on the flow distribution and typical velocities in the break stage, including jetting on the shroud, the far field and the upper separator and dryer regions.
- The aerosol size distribution at the inlet and outlet of the ARTIST facility, and possible shifting of the distribution.
- The inlet and outlet aerosol concentration of each stage and corresponding DFs.
- Aerosol mass balance in the test section, including deposition fractions in the break stage, the rest of the bundle, and the separator and dryer regions.

8. Summary and current status

An experimental facility (ARTIST) to investigate aerosol and iodine retention in a PWR SG following an SGTR has been designed. The facility is a scaled representation of the Framatome designed SG in the Beznau plant (KKB). The construction is currently in progress and will be complete in the early part of 2000. The experimental programme will provide data to address a range of issues. These include quantifying retention in a dry SG during a severe accident, AM measures to refill the SG, and iodine retention in the SG following a design basis accident.

Planning of the test matrix is in the early stages. A postulated reference severe accident SGTR sequence in KKB has been analysed and used to define a baseline set of conditions as a starting point for the ARTIST experiments. Based on the analysis, the various aerosol retention mechanisms have been discussed and broadly quantified for the prevailing conditions. Scaling considerations show that different, indeed highly contrasting scale factors apply to capture the different mechanisms that take place in the various regions of the SG; in particular, different scaling principles apply locally near the break and globally. Test conditions have been defined corresponding to the reference case.

The baseline conditions apply to a particular plant and set of accident conditions; however, a range of plant designs and accident conditions can be accommodated by suitably defining the experimental conditions, via analogous scaling arguments. The Paul Scherrer Institut is pleased to offer its capability to address SG retention issues of concern to industry and regulators in the wider international nuclear community.

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References

1. Excerpt from KKB PSA Study (private communication).
2. C. M. Allison et al, "SCDAP/RELAP5/ MOD3.1 Code Manual", NUREG/CR-6150, June 1995.
3. J.C. Birchley "Reference Calculation for STGR Experiments: Aerosol Trapping in Steam Generator (ARTIST) Program", ARTIST-01/99, September 1999.
4. "Technical Base for Estimating Fission Product Behaviour during LWR Accidents", NUREG-0772.
5. B.Y.H. Liu and J.K. Agarwal, "Exper-imental Observation of Aerosol Deposition in Turbulent Flow", J. Aerosol Sci., Vol 5, pp. 145-155, 1974
6. P. Douglas and S. Ilias, "On the Deposition of Aerosol Particles on Cylinders in Turbulent Cross Flow", J. Aerosol Sci., Vol 19, No.4, pp. 451-462, 1988.
7. J. Quarini, "CFD Model of the PSI Steam Generator", Report submitted to PSI, January 1999.
8. Y. Ye and D.Y.H Pui, "Particle Deposition in a Tube with and Abrupt Contraction", J. Aerosol Sci., Vol 21, No.1, pp. 29-40, 1990.
9. W. Hinds, Aerosol Technology, John Wiley & Sons, 1982.
10. Dehbi, D. Suckow, S. Güntay, "Aerosol Retention of Hot Pools under Realistic Accident Conditions", OECD-CSNI Meeting, June 1998.
11. J.P. Adams, "Iodine Release During a Steam Generator Tube Rupture" Nuclear Technology, vol 109, Feb. 1995.
12. K. Garbett et al, "Coincident Steam Generator Tube Rupture and Stuck-Open Safety Relief Valve Carryover Tests", NUREG/CR-4752, EPRI NP-4787, TPRD/L/3009/R86, WCAP-11226, March 1987.

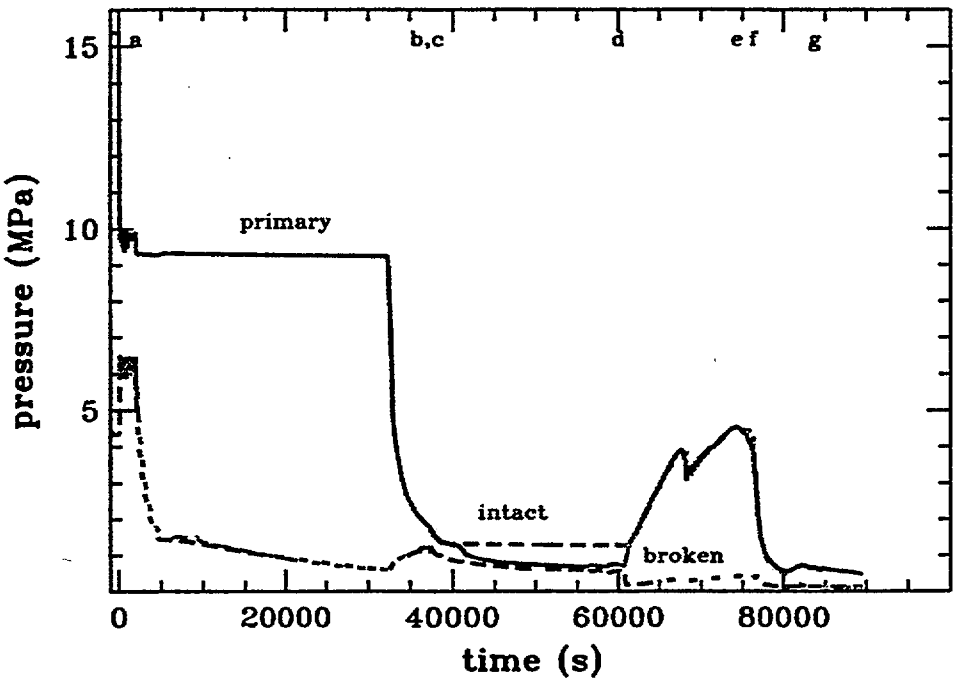


Figure 1: Primary and SG secondary side pressures

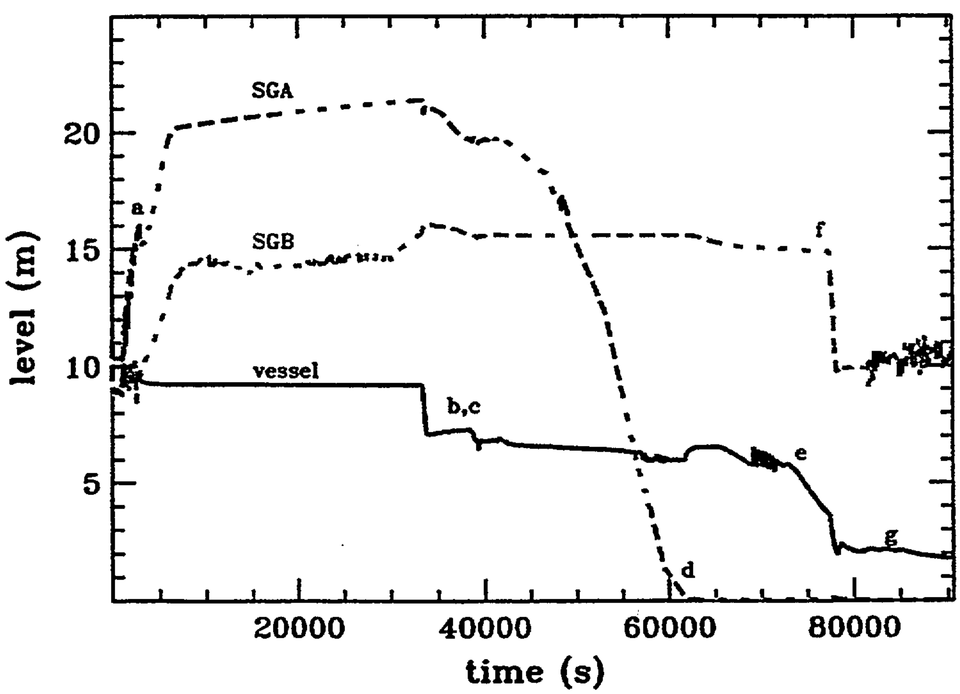


Figure 2: Reactor vessel and SG secondary liquid levels

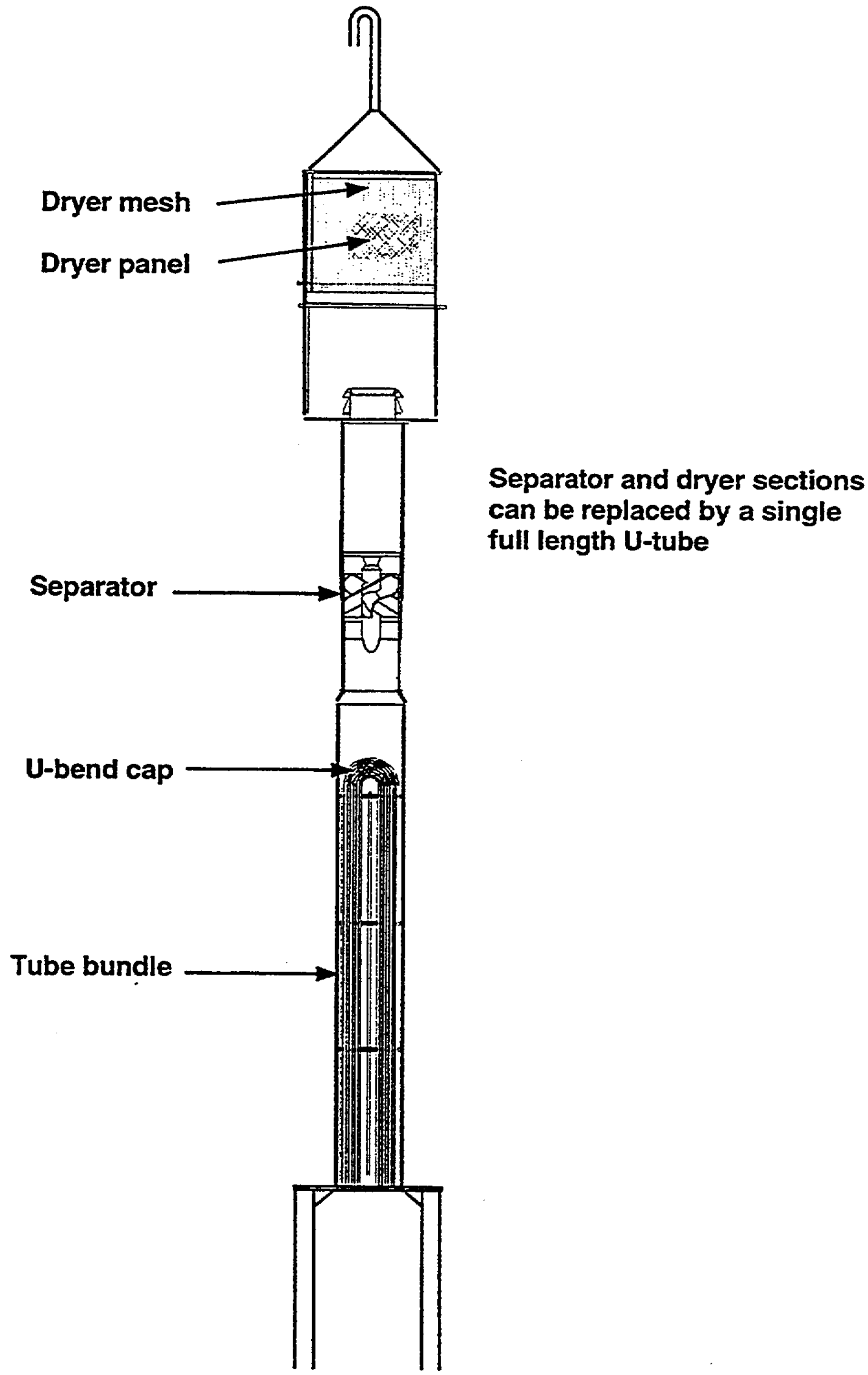


Figure 3: Schematic of ARTIST facility (to scale)

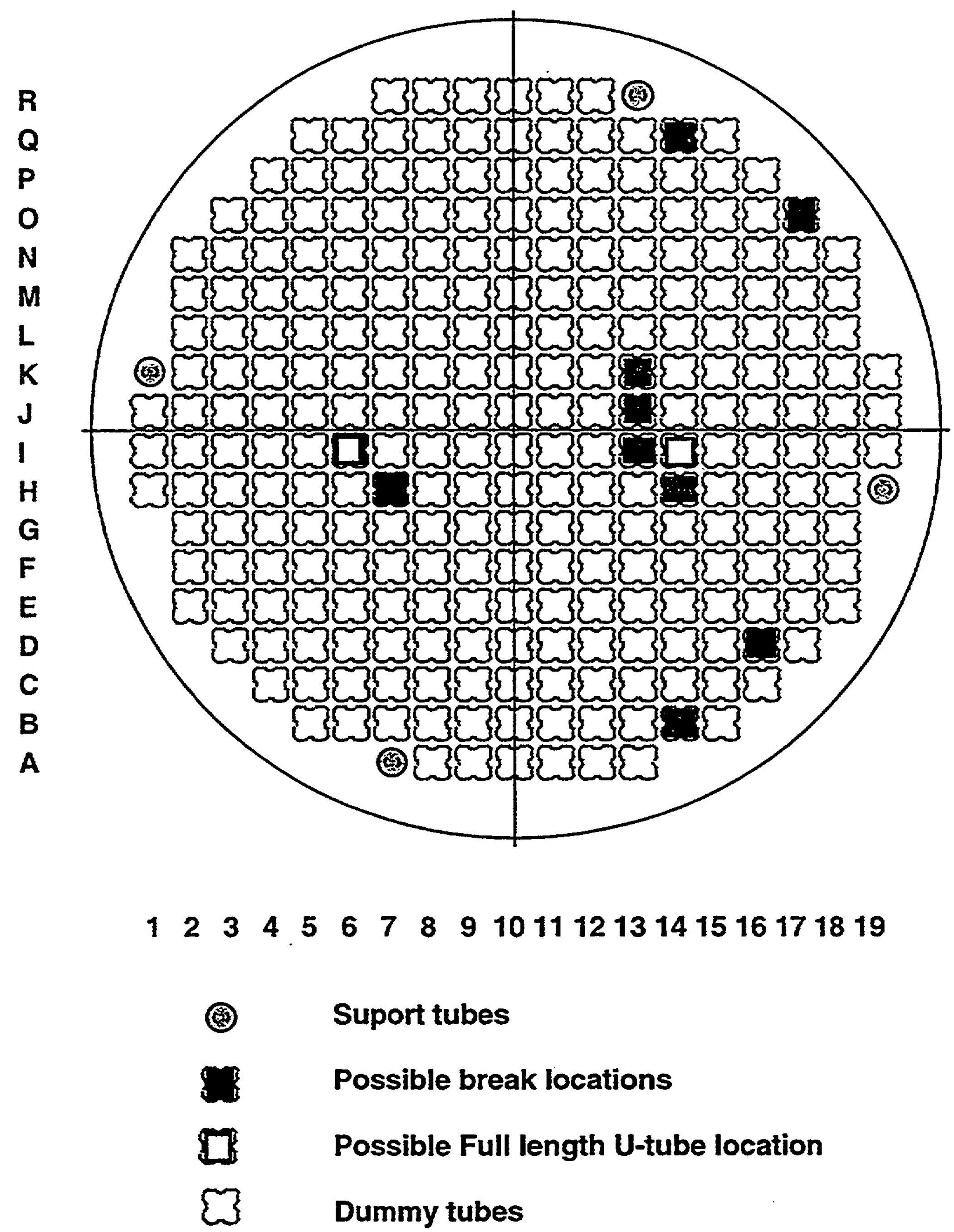


Figure 4: ARTIST support plate, tube layout and break locations

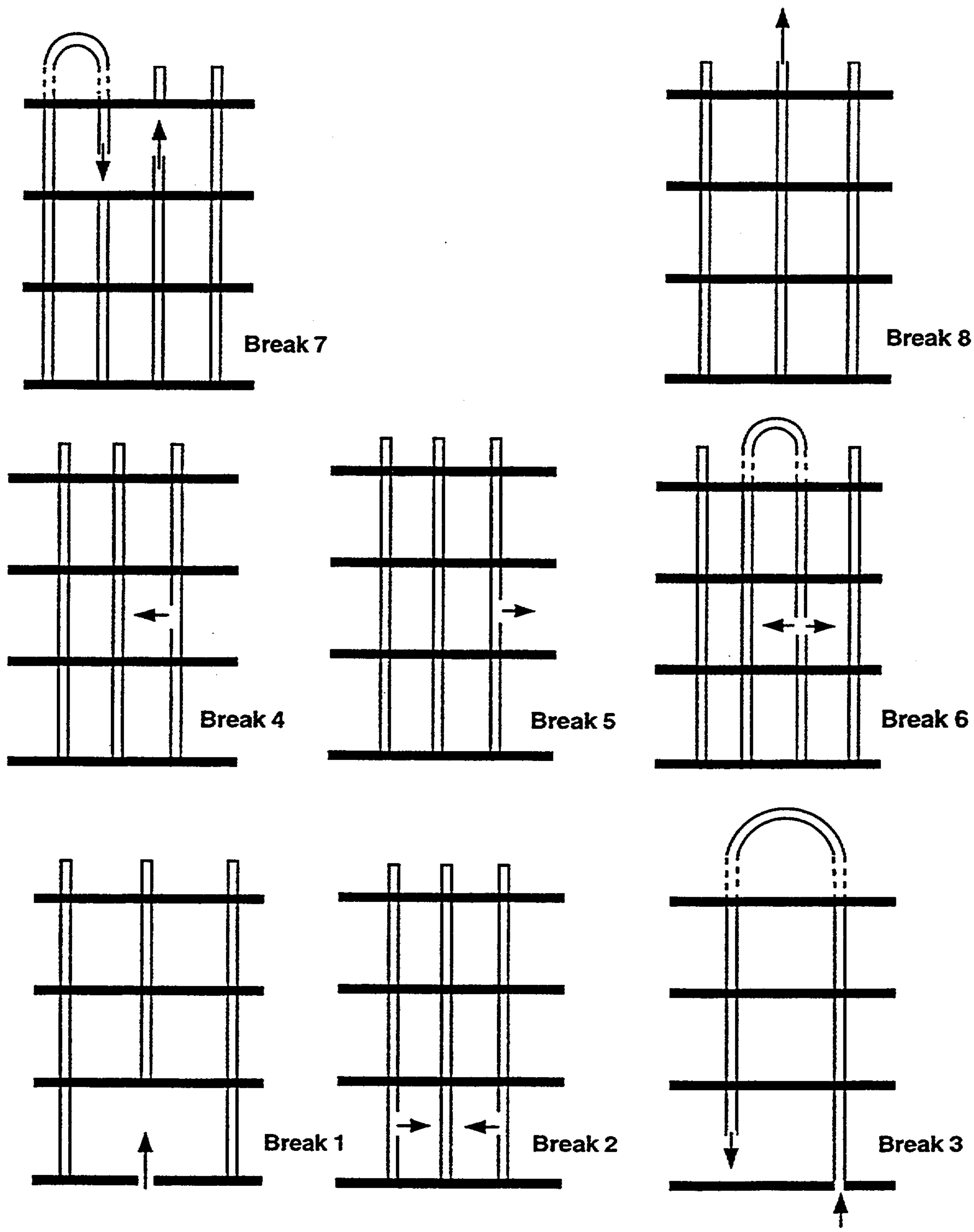


Figure 5: Break Locations in ARTIST bundle

P= pressure
T= gas temperature
T_w = wall temperature
Imp=impactor
Fil= membrane filter
Ph= photometer
F= Flowmeter
An= anemometer
Dc= deposition coupon
Gs= grab sample
Q= QCM impactor

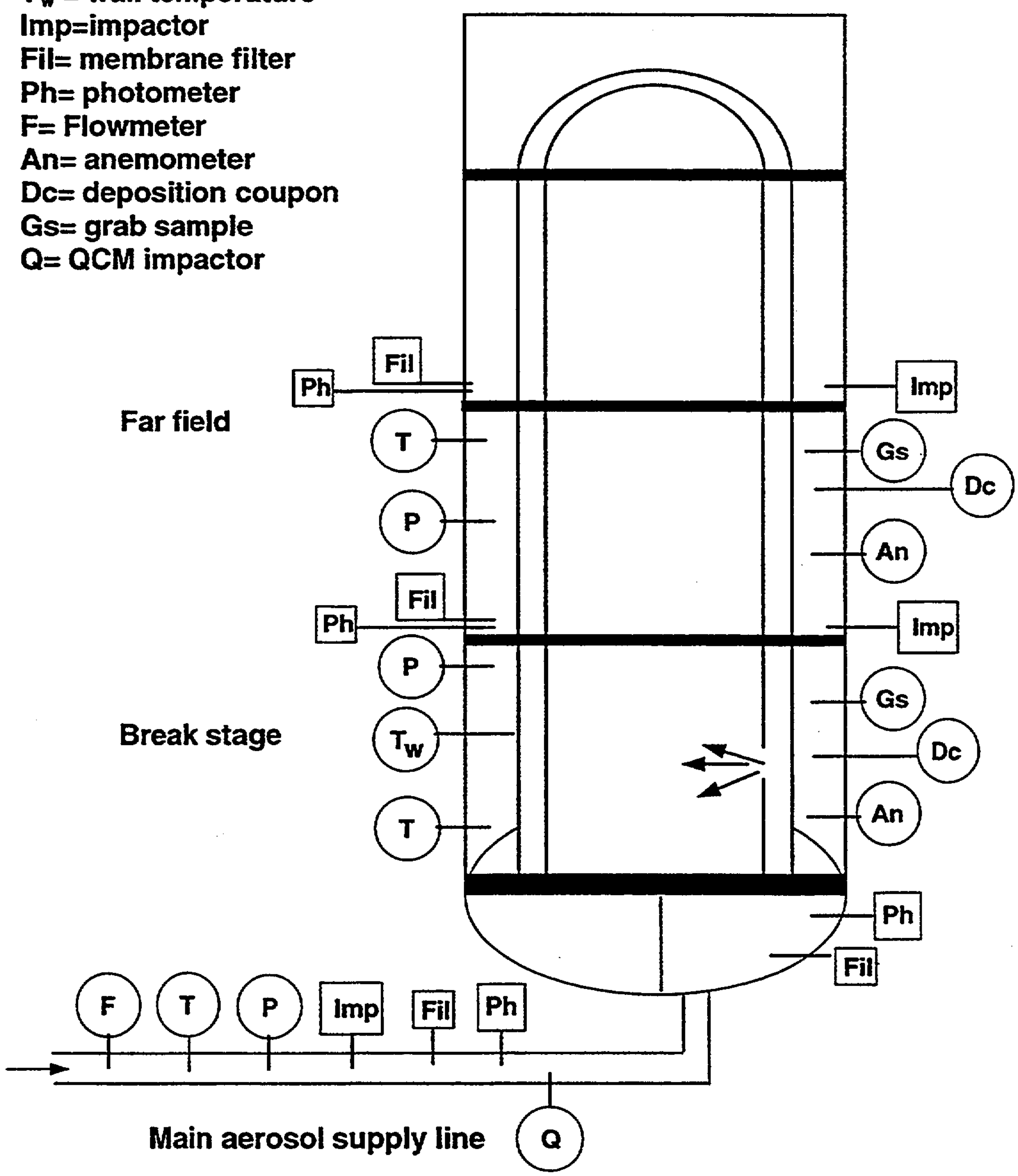
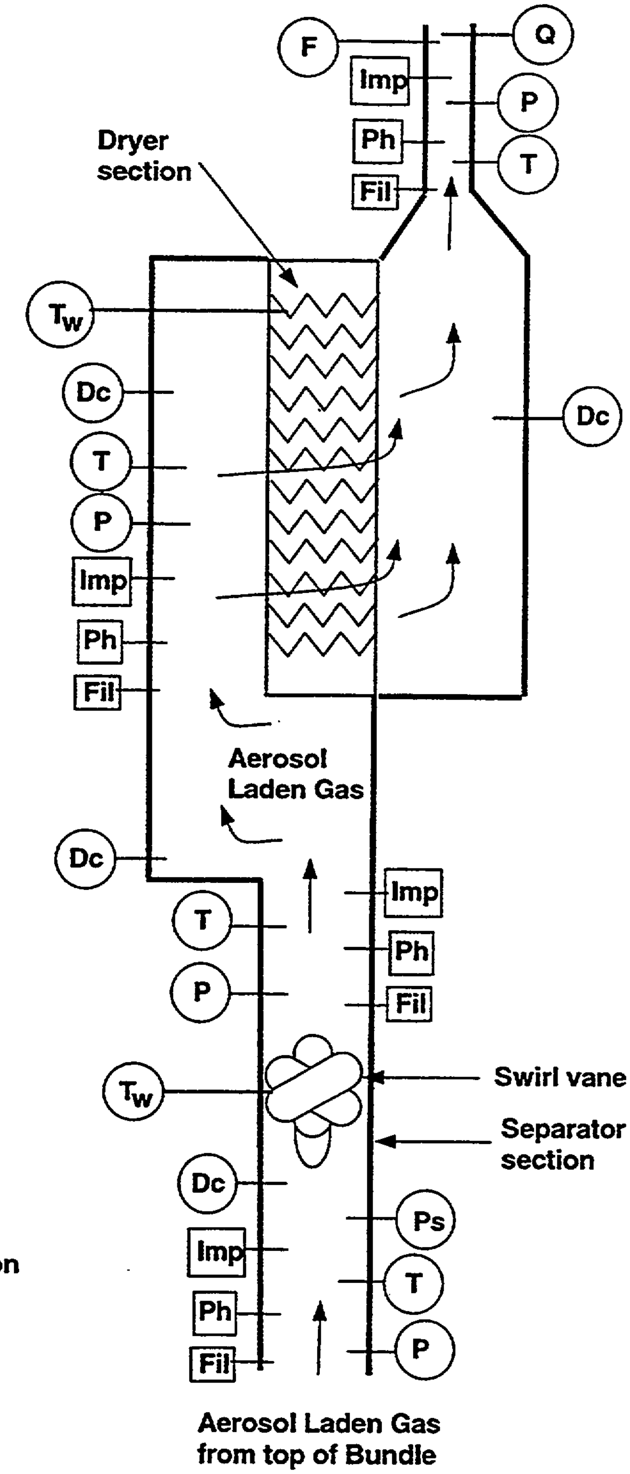


Figure 6: Schematic of preliminary instrumentation in inlet line, inlet plenum, break stage and far field stage



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Dc= deposition coupon
Gs= grab sample
Ps=Psychrometer
Q=QCM impactor

Figure 7: Schematic of preliminary instrumentation in separator and dryer

Digital Systems Safety Assessment

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Abstract

Computer systems are increasingly being used in life-critical applications such as flight control, nuclear reactors, railway applications and medical devices. Designers need techniques to help ensure that proper design decisions are made during the design process, and regulatory bodies need approaches that can be used to assess the safety of systems before they are allowed to operate in the field. Typically, Markov models, fault trees, flow graphs, and/or Petri nets are used to model the high-level behavior of safety-critical systems, and techniques and tools to develop such safety models are well established. However, the safety models contain certain parameters that are often very difficult to estimate. Examples of such parameters are failure rates and conditional probabilities such as fault coverage. This paper presents an assessment methodology that can be applied to safety-critical applications. The methodology considers permanent operational faults, transient operational faults, and design faults and uses hardware/software simulation as a method of estimating crucial parameters such as fault coverage. As a proof of concept, the methodology was applied to an existing Interlocking Control System (ICS).

1. Introduction

The use of computers to control life-critical systems is becoming more prevalent with the availability of low-cost, high-performance microprocessors. One of the difficulties in using a computer system in a life-critical application is verifying that the system is safe. Digital systems, like all systems, are susceptible to failures caused by defects within the system. Typically, defects are referred to as faults and are introduced into the system either during the system design or during the operation of system. These two categories of faults are commonly referred to as design faults and operational faults, respectively. For safe operation to be achieved, the system must be designed to detect a large percentage of operational and design faults. The conditional probability of detecting a fault given that a fault is in the system is referred to as *fault detection coverage*. For a system to be considered safe, the designer must demonstrate that the fault detection coverage of the system is very close to 1.0. Estimating fault detection coverage is a difficult and resource intensive task [1].

Another parameter which affects safety is the fault arrival rate associated with the system. The rate at which faults arrive is a function of component wear-out and transient disturbances. The component wear-out aspect is classified as a permanent fault while transient disturbances are categorized as transient faults. The fault arrival rate for permanent faults is typically referred to as a failure rate. The failure rate of a hardware device is typically obtained from an analytical model such as MIL-HBK-217 [2]. The transient fault arrival rate is typically estimated by noting that transient faults account for 80% to 90% of all system failures [3, 4, 5, 6, 7]. Thus, the transient failure rate can be estimated from the permanent failure rate. Unfortunately, there are no similar, existing techniques to estimate arrival rates for design faults, and other approaches must be used.

This paper presents a methodology for assessing the safety of critical systems. Specifically, a methodology was developed to estimate a system's Mean Time Between Hazardous Events (MTBHE) [8, 9]. The example system used to illustrate the proposed methodology is an Interlocking Control System (ICS) that is used for railroad signalling and switch control. Specifically, the ICS is used to control the switches and signal lights for a given section of track.

The methodology consists of three major components: (1) the construction of a safety model by representing high-level system behavior with a Petri net model; (2) the estimation of the failure rates contained in the safety model and the estimation of the fault coverage of the system; and (3) the solution of the safety model for the calculation of the safety metrics. This methodology is concerned with permanent, transient, and design faults. Most safety models described in literature are concerned solely with permanent faults. Additionally, the fault coverage of the system is divided into fault coverage for permanent faults and fault coverage for transient faults. Several unique techniques were developed to assist the fault coverage estimation process. Specifically, the concept of fault expansion [10] and variance reduction based on fault expansion [11] are employed.

The major contribution of this paper is the presentation of a methodology which has been used to estimate the safety of an ultra-safe system. Simulation of the complete hardware/software system plays a major role in the assessment methodology. Transient, permanent operational, and design faults are considered with the presented safety verification methodology.

This paper is organized in the following manner. Following this brief introduction the overall safety assessment process is presented. Next, a description of the example interlocking control system (ICS) that has been modeled is presented. This is followed by a description of the parameter estimation techniques used to estimate failure rates and fault coverage. Results associated with the assessment of the ICS are included. Finally, concluding remarks are provided.

2. Safety assessment process

Before the details associated with the safety evaluation methodology are presented some background associated with safety metrics is introduced. The first metric is referred to as safety and is defined as the probability of the system operating in a safe fashion over a time interval given that the system is in a safe state at the beginning of the time interval [2]. Typically, safety is represented mathematically by the function $s(t)$ where t is time. For most real world systems it is assumed that the safety function $s(t)$ starts at time $t = 0$ with a value of 1.0. As time increases the safety function decreases to zero in the limit as time goes to infinity.

Another common safety metric is the Mean Time Between Hazardous Events (MTBHE). The MTBHE metric estimates the average time between unsafe system failures. Conceptually, the MTBHE is the expected time before a given system fails in an unsafe manner. The MTBHE specification is typically given as the average number of years or hours of safe operation before an unsafe failure occurs.

The safety evaluation methodology can be thought of as a modeling process which consists of three levels of models -- axiomatic models, empirical models, and physical models. The interrelationship of the layers of modeling is presented graphically by Figure 1. The highest level of modeling is an axiomatic model. For the safety evaluation methodology the axiomatic models are analytical models that are used to model the structure and the safety behavior of a system. The axiomatic model can be used to estimate $s(t)$ or MTBHE for a given system. Markov models, Petri nets, and fault trees are examples of axiomatic models.

Deriving and solving Markov and/or Petri net models for most safety architectures is a straightforward process so long as one can determine the required safety parameters. The failure rate, repair rate, and fault coverage are all examples of safety parameters. The estimation of safety metrics is now reduced to accurately estimating the safety parameters associated with a given system. Once these parameters are known, then the safety metrics of interest can be readily calculated from the probability functions associated with each state of the Markov model.

Axiomatic models contain parameters which must be estimated before the solution to the model can be obtained. Failure rates, repair rates, and fault coverage are all examples of axiomatic model parameters. The failure rate of a component is simply the expected number of failures per unit time. For example if a component on average fails once every 10,000 hours then the failure rate is 1 failure per 10,000 hours or simply 10^{-4} failures per hour. Likewise, the repair rate of a component is the average number of repairs which are performed per unit time. The probability of detecting and properly processing a system fault is referred to as fault coverage. Mathematically, fault coverage is defined to be the conditional probability of detecting and properly processing a fault given that a fault exists in the system [2].

Empirical models are the second level of modeling associated with the safety evaluation methodology. Empirical models can be used to provide estimates of axiomatic model parameters. An empirical model is a statistical model that is used to represent complex and detailed descriptions of the system parameters using collected data. Specifically, experiments are performed on either simulation models or prototypes to generate data. The empirical model then processes the data to generate the estimate of the axiomatic model parameters. The simulation models or system prototypes are referred to as physical models. Physical models are prototypes that implement the hardware and/or software of an actual system. Physical models are the third layer of modeling support in the safety evaluation methodology.

The parameter values used are approximations derived from expert opinion, derived from other high-level analytical models, or they are estimated using empirical and physical models. If statistical estimation

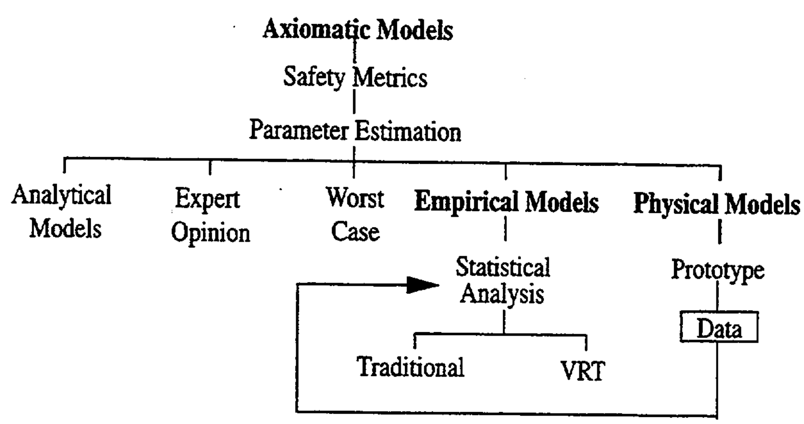

Figure 1. Safety assessment methodology [12].

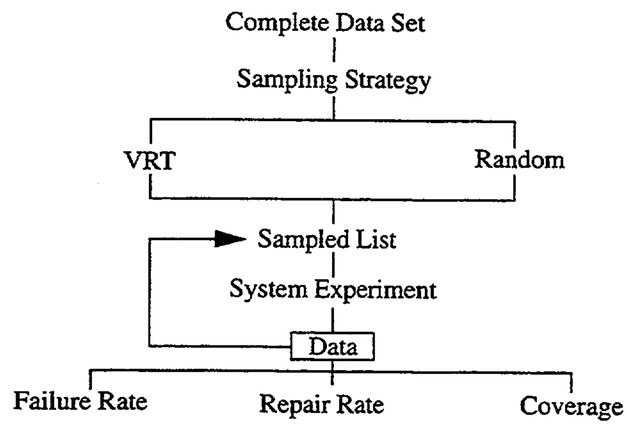

Figure 2. Parameter estimation approach [12].

is employed, then empirical models derive estimates for these parameters from data collected from physical models to use in calculating safety metrics. The process used for parameter estimation is shown in Figure 2. The top level of this model represents the complete set of possible data values which are to be sampled. A sampling strategy is then determined. Random sampling, stratified sampling, and importance sampling are examples of different sampling strategies which can be employed. Importance sampling and stratified sampling are examples of Variance Reduction Techniques (VRT). The objective of VRT is to increase the accuracy of the estimate by exploiting some known attribute of the system. The selected sampling strategy is then employed to generate a sampled list. The physical model is utilized to evaluate the sampled list using system experiments. The observed data is evaluated using the empirical model and the estimate for the parameter of interest such as failure rate, repair rate, or fault coverage is obtained.

The failure rate of a system component can be estimated by either analytical or statistical techniques. The analytical techniques are based on failure rate models which are derived from analyzing the failures associated with actual devices. The most common analytical technique is the United States Department of Defense (USDOD) MIL-HBK-217 standard [2]. The failure rate of a component can also be determined using statistical techniques. Typically, this statistical estimate is obtained after a large number of systems have been in service over a period of years. This approach is only feasible in a limited number of applications. For example, the phone company typically collects this failure rate information for the various phone switching systems [13]. Additionally, in some cases expert opinion is used to estimate the failure rate information.

Of all the axiomatic parameters, estimating the fault coverage of a system is the most difficult. This difficulty is caused in most cases by the lack of analytical estimation methods. Specifically, the fault coverage is almost always estimated using a Monte Carlo statistical approach [1]. Data is collected by a three step process; (1) a fault of interest is selected, (2) the fault is introduced into a prototype/simulated system using fault injection [4, 14-18], and (3) the system is monitored to determine if the injected fault is properly processed.

The injected faults are typically selected at random from the entire system fault space. This approach requires a huge number of fault injection experiments to accurately estimate coverage for an ultra-safe and/or ultra-reliable system. Compounding this fault coverage estimation process is the fact that a percentage of these randomly selected faults typically remain latent for an extremely long duration; that is, the faults never become active and produce an error. These latent (no response) fault injection experiments effectively provide little if any information about fault coverage. Thus, depending on the percentage of no response faults,

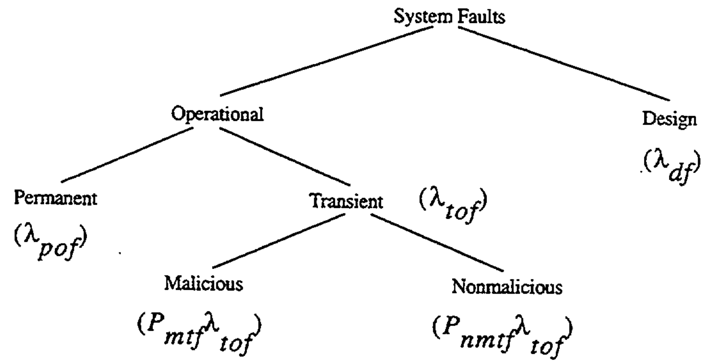

Figure 3. Categories of faults considered.

significantly more randomly selected fault injection experiments are required to collect the data points needed to accurately estimate a fault coverage.

Another seldom-used technique for estimating coverage is to analyze the detection mechanisms, develop a detection model, and solve the model for coverage. Unfortunately very few detection mechanisms exist which lend themselves to this form of analysis. The only known class of detection mechanisms which are analyzed in this fashion is concurrent error detection techniques. One such concurrent error detection scheme that fits this category and has been used for safety-critical train control is presented by Smith et al [19]. Analytically estimating the coverage value is acceptable so long as reasonable assumptions are made in the modeling and analysis of the detection mechanism.

3. Example system assessment

The first step in the methodology is the construction of a safety axiomatic model for the system of interest. The structure of the safety axiomatic model is tightly coupled to the structure of the system and the manner in which faults are categorized. The types of faults considered by this safety verification technique are first presented followed by the structure of the safety model for the example ICS.

The categories of faults considered for the safety evaluation of the ICS can be divided into two broad categories: (1) the design faults where the failure rate is given as λ_{df} ; and (2) the operational faults. The design fault category includes both hardware and software design faults. The operational faults are further divided into permanent operational faults and transient faults. The failure rates for permanent and transient faults are given as λ_{pof} and λ_{tof}, respectively. The transient faults are further subdivided into malicious and nonmalicious categories. The conditional probability that a transient fault is malicious or nonmalicious given a transient fault has occurred is represented by P_{mtf} and P_{nmtf} , respectively. A fault is considered to be malicious if it will cause a system failure when all of the fault detection mechanisms of the system are deactivated. An algorithm is used to locate malicious transient faults which can occur [20]. A graphical depiction of how the fault categories are related is included as Figure 3.

The ICS that is used to demonstrate the methodology is a fail-safe programmable logic controller with a simplex architecture. An 8-bit microprocessor executes software which periodically reads the system inputs, processes the user defined interlocking control equations, and delivers the calculated outputs. Several fault detection mechanisms are employed to achieve system safety. Software diagnostics are run extensively to detect operational and design faults. A watchdog timer is also employed to assure that the 8-bit microprocessor is still executing the diagnostic software and control equations at a predefined rate. The ICS requires that all portions of the system be fault free for the system to remain in a safe operational state. The safety requirement for the ICS is that the MTBHE must be equal to or greater than 10^7 years.

A *Generalized Stochastic Petri Net* (GSPN) is used to represent the high-level system behavior. A Petri Net (PN) [21] is a bipartite directed graph whose nodes are divided into two disjoint sets called *places* and *transitions*. Directed arcs in the graph are drawn only from places to transitions (called input arcs) and from transitions to places (called output arcs). A *Marked Petri net* is obtained by associating *tokens* with places. In a graphical representation of a PN, places are represented by circles, transitions are represented by bars and the tokens are represented by black dots in the place or a number denoting the number of tokens in that place. In typical Petri Net models, places represent conditions while transitions represent events. Places may also represent resources, and the number of tokens in a place may denote the number of redundant resources available of a certain type. Transitions may also indicate different choices of behavior of a system.

A GSPN can be solved directly or converted to a Continuous-time Markov Chain (CTMC) [22]. The approach taken with the presented methodology is the conversion of the GSPN into a CTMC. A graphical depiction of the Petri net safety model is included as Figure 4. The derivation of the Petri net begins by assuming the ICS is in a *System Operational Place*. A token can flow from the *System Operational Place* to either the *Permanent Fault Place* or the *Transient Fault Place* through one of the three timed transitions. These transitions have a rate associated with them which are the rate of permanent faults and the rate of transient faults (λ_{tof}) that occur in the system. The rate of permanent faults is divided into two types: permanent operational faults (λ_{pof}) and design faults (λ_{df}). Given that a transient fault has occurred, the token may then flow into the *Malicious Transient Fault Place* with probability P_{mtf}, or back to the *System Operational Place* (if the fault is not malicious) with probability $1-P_{mtf}$. Given that a malicious transient fault has occurred, the token can then flow either to the *Unsafe Place* with probability of $1-P_{mtfd}$, or back to the *Operational Place* with probability P_{mtfd}. Given that a permanent fault has occurred, the token can either flow to the *Safe Place* with probability P_{pfd}, or to the *Unsafe Place* with probability $1-P_{pfd}$.

In deriving this model, it is assumed that design faults will be processed by the ICS in a similar fashion as permanent operational faults. This assumption is reasonable because design faults can be modeled as latent permanent operational faults which are present in the system at time $t = 0$.

In the Petri net, the places denoted *Safe* and *Unsafe* are absorbing places since a token may flow into but not out of those places. This absorption occurs because the ICS is a fail-safe system. When a token flows into the *Unsafe Place* it indicates that an unsafe failure state has been reached. For convenience, the fault detection probabilities of the Safety Petri net are summarized below.

- P_{mtfd} - Probability of detected malicious fault/error
- P_{mtfud} - Probability of undetected malicious fault/error ($P_{mtfud} = 1-P_{mtfd}$)
- P_{pfd} - Probability of detected permanent fault
- P_{pfud} - Probability of undetected permanent fault ($P_{pfud} = 1-P_{pfd}$)

Likewise, the failure rates for the Safety Petri net are depicted in Figure 4.

4. Parameter Estimation

The analytical expression to represent safety for most safety models contains two types of parameters: (1) failure and repair rates, and (2) conditional probabilities such as fault detection coverage. Because the ICS is a simplex system, the safety model does not contain repair rate parameters. The MTBHE expression contains failure rate parameters, fault detection parameters, and a probability of a malicious fault parameter. This paper focuses on the determination of the coverage parameter using fault simulation.

The coverage estimation of permanent and transient fault detection coverages is performed using a single statistical estimation techniques. Because the safety of a system is highly sensitive to slight variations of coverage [8], extreme care must be taken when estimating coverage. Typically, coverage is estimated statistically and a confidence interval is derived to indicate the accuracy of the estimate [1]. The major difficulty with traditional approaches is the large amount of data required for coverages which are very close to 1.0. For the ICS verification effort it was determined that the coverage estimate would have to exceed 0.9_6 for the MTBHE specification to be satisfied. More than 100,000 data points are required to demonstrate this level of coverage.

One way to decrease the required number of data points is to increase the accuracy of the estimate by exploiting some attribute of the system. Reducing the variance associated with the estimate is one way in which the accuracy of the estimate can be increased [24, 25]. A novel Variance Reduction Technique (VRT) is employed to assist in increasing the accuracy of the coverage estimate. The low-level details of the VRT are described in [11].

The transient fault detection coverage is estimated by sampling transient faults that will cause the system to fail if they are not detected. Faults which have this property are referred to as malicious. An algorithm which locates malicious transient faults by analyzing the low-level information flow through the system under test [20] is known as the Malicious Fault List Generation Algorithm (MFLGA) and enumerates all of the malicious transient faults which cause a corruption in data processing.

5. Results

The results for the safety assessment of the example ICS fall into two broad categories: (1) parameter estimates and (2) safety metrics, such as MTBHE. The estimation of the permanent operational failure rate

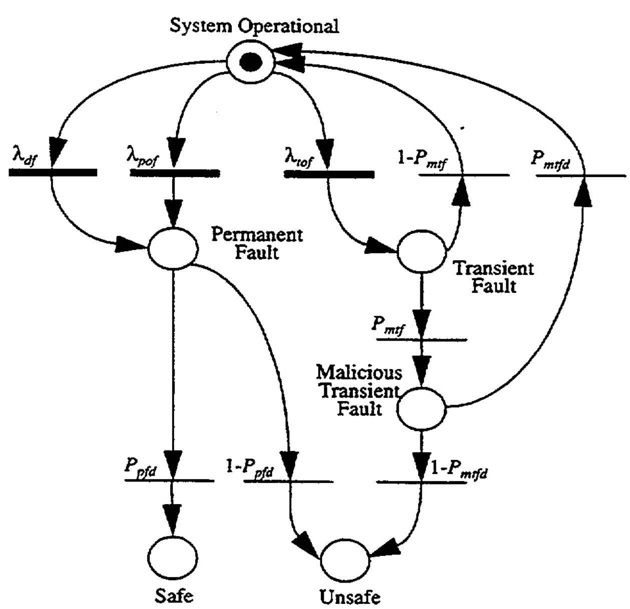

Figure 4. ICS safety Petri net model.

for the ICS is accomplished in two ways; that is, statistical estimation and MIL-HBK-217. The estimated and calculated failure rates are depicted in the first and second columns of Table 1.

Field Failure Rate (90% CL), faults/year	Predicted Failure Rate, faults/year
0.2306	4.3031

TABLE 1. ICS hardware failure rates.

The permanent and transient operational fault detection coverage estimates were obtained by running fault injection experiments on a model of the ICS. Specifically, the ICS was modeled using the ISA modeling methodology [26]. The ICS processor (Motorola 6809 microprocessor) was modeled using the Very high speed integrated circuit Hardware Description Language (VHDL). The 6809 ISA model executes machine language instructions with the appropriate timing characteristics as specified by the Motorola processor handbook [26, 27]. The rest of the system peripheral hardware was modeled using a behavioral VHDL description. The actual ICS software used in the field was executed on this model.

The evaluated permanent faults consist of single stuck-at faults in the memory and registers of the ICS. Likewise, the evaluated transient faults consist of a set of malicious transient faults generated by the MFLGA. Fault expansion was then applied to the injected faults by the technique described in [10]. The simulation results are:

n_{pf} = # injected permanent faults = 100,522

n_{tf} = # injected transient faults = 1,849

m_{pf} = # expanded permanent faults = 9,410,793,764

m_{tf} = # expanded transient faults = 136,494,081

Using these quantities, the coverage values for permanent faults (P_{pfd}) and malicious transient faults (P_{mtfd}) were calculated. The results are shown in Table 2.

Coverage for Permanent Faults, P_{pfd}	Coverage for Malicious Transient Faults, P_{mtfd}
0.9_7465	0.9_56726

TABLE 2. Coverage estimates (at 90% C.L.)

The conditional probability of a malicious transient fault given that a transient fault has occurred is calculated by enumerating the number of malicious transient faults (η_{mtf}) and the number of transient faults η_{tf} for a given time interval. To provide a worst case estimate of P_{mtf}, the evaluated time interval is selected when the ICS is processing safety critical data. Selecting the time interval in this fashion will cause the estimate for (η_{mtf}) to be either accurate or too large, which results in a conservative upper bound estimate of P_{mtf}.

The combined failure rate for an individual ICS unit is shown in Table 3. Using the failure rates listed in this table, the P_{mtf} estimate and the coverage estimates provided in Table 2, the MTBHE for the ICS can be calculated. The results using both the predicted and field-demonstrated failure rates are shown in Table 3. As can be seen from the table, the MTBHE using field-demonstrated failure rates exceeds the MTBHE requirement of 10^7 years. The use of the field failure rates is strongly supported by the fact that the ICS has a long and successful field history; that is, the ICS has performed several million hours of safe operation.

6. Conclusions

A methodology for the assessment of safety-critical systems is presented. The methodology uses permanent operational faults, transient operational faults, and design faults to calculate safety. The technique consists of a three step process: (1) developing a safety model and solving the model to provide an analytical expression for safety; (2) estimating parameters required by the analytical safety expression; and (3) evaluating the analytical safety expression.

The most costly portion of the process concerns the accurate estimation of the safety model parameters. Typically, the most difficult parameter to estimate is fault detection coverage. A novel VRT which uses equivalent fault sets is employed to increase the accuracy of the fault detection coverage estimates [11]. The presented VRT can significantly reduce the number of data points required to estimate fault detection coverage. Another novel aspect of the presented methodology is the use of a MFLGA to estimate the transient fault detection coverage. The MFLGA presented in [20] determines which transient faults produce an error, and if they are undetected will cause a system failure. Randomly sampled transient faults do not have this attribute, and for this reason have a high probability of not causing any discernible effect on the system under evaluation. The other additional benefit of the MFLGA is the ability to estimate the conditional probability of malicious transient fault given a transient fault has occurred. This conditional malicious transient fault probability is then incorporated into the safety model to increase the accuracy of the model.

The derived safety model has two unique attributes: 1) both operational and design faults are considered and 2) the system is modeled as combined hardware/software entity. Most evaluation techniques simply consider operational faults. For safety-critical applications both design and operational faults should be considered. The use of a combined software/hardware model provides greater accuracy than the common practice of analyzing the hardware and software independently. Specifically, hardware faults can cause the software to behave in unexpected ways and conversely software faults can cause the hardware to react in unexpected ways. The independent evalution of hardware and software ignores the interaction between the two domains.

The presented technique was applied to a railroad ICS which has several million hours of safe operation. The objective was to demonstrate that the ICS has a MTBHE of at least10^7 years. The ICS was modeled using an ISA VHDL model. Fault injection experiments were then performed on the ISA model to obtain the data necessary to estimate the fault detection coverages. The safety model for the ICS was then derived and solved. The end result of the ICS verification effort is an MTBHE estimate of 7.72×10^7 years with 90% confidence level. Thus, the developed methodology can be used to demonstrate extremely high-levels of safety.

Predicted Failure Rates	**Field-Demonstrated Failure Rates**
MTBHE, years	MTBHE, years
4.13×10^6	$\mathbf{7.71 \times 10^7}$

TABLE 3. ICS MTBHE (at 90% C.L.)

7. References

[1] D. Powell, E. Martins, J. Arlat, "Estimators for Fault Tolerance Coverage Evaluation", *23rd International Symp. on Fault-Tolerant Computing*, June 1993, pp. 228-237.

[2] B. W. Johnson, *Design and Analysis of Fault-Tolerant Digital Systems,* Addison-Wesley, Massachusetts, 1989

[3] M. Ball and F. Hardie, "Effects and Detection of Intermittent Failures in Digital Systems," *AFIPS Fall Joint Computer Conf. Proc*, pp. 329-335, 1969.

[4] R. K. Iyer and D. J. Rossetti, "A Measurement-Based Model for Workload Dependence of CPU Errors, "*IEEE Transactions on Computers*, Vol. C-35, pp. 511-519, June 1986.

[5] R. K. Iyer, D. J. Rossetti, and M. C. Hsueh, "Measurement and Modeling of Computer Reliability as Affected by System Activity," *ACM Transactions on Computer Systems*, Vol. 4, No. 3, August 1986, pp. 214-237.

[6] R. K. Iyer and D. J. Rossetti, "A Measurement-Based Model for Workload Dependence of CPU Errors," *IEEE Transactions on Computers*, Vol. C-35, June 1986, pp. 511-519.

[7] R. K. Iyer and P. Velardi, "Hardware-Related Software Errors: Measurement and Analysis," *IEEE Transactions on Software Engineering.*, Vol. SE-11, No. 2, Feb. 1985, pp. 223-231.

[8] C. Choi, B. Johnson, J. Profeta, *Safety Issues in the Comparative Analysis of Dependable Architectures,* Technical Report TR1222.0, Center for Semicustom Integrated Systems, Electrical Engineering Department, University of Virginia, Dec. 1994.

[9] H. Choi, W. Wang, and K.S. Trivedi, "Conditional MTTF and its Computation in Markov Reliability Models," *Proc. 1993 Annual Reliability and Maintainability Symposium*, January 25-28, 1993, pp. 55-63.

[10] D. T. Smith, B. W. Johnson, J. Profeta, and D. Bozzolo, "A Method to Determine Equivalent Fault Classes for Permanent and Transient Faults", *Reliability and Maintainability Symposium*,1995, pp. 418-424.

[11] D. T. Smith, B. W. Johnson, N. Andrianos, and J. Profeta, "A Variance-Reduction Technique via Fault-Expansion for Fault-Coverage Estimation", *IEEE Transactions on Reliability,* Sept. 1997, Vol. 46, No. 3, pp. 366-374.

[12] L. M. Kaufman, "Dependability Analysis for Ultra-Dependable Systems Using Statistics of the Extremes," Ph.D. Dissertation, University of Virginia, 1997.

[13] W. N. Troy, "Fault-Tolerant Design of Local ESS Processors", *Proceedings of the IEEE*, October 1978, pp. 1126-1145.

[14] Z. Segall, D. Vrsalovic, D. Siewiorek, D. Yaskin, J. Kownacki, J. Barton, R. Dancey, A. Robinson, and T. Lin, "FIAT: Fault Injection Based Automated Testing Environment", *18th International Symposium on Fault-Tolerant Computing*, pp. 102-107.

[15] Ghani A. Kanawati, Nasser A. Kanawati, and Jacob A. Abraham, "FERRARI: A Tool for The Validation of System Dependability Properties", *22nd International Symposium on Fault-Tolerant Computing*, July 1992, pp. 336-344.

[16] John G. McGough, and Salvatore J. Bavuso, "Methodology for Measurement of Fault Latency in a Digital Avionic Miniprocessor", *4th AIAA/IEEE Digital Avionics Systems Conference*, 1981, pp. 301-314.

[17] Algirdas Avizienis and David A. Rennels, "Fault-Tolerance Experiments with the JPL STAR Computer", *COMPCON*, September 1972, pp. 321-324.

[18] Aamer Mahmood, and E. J. McCluskey, "Concurrent Error Detection Using Watchdog Processors - A Survey", *IEEE Transactions on Computers*, Vol. 37, No. 2, February 1988, pp. 160-174.

[19] D. T. Smith, B. W. Johnson, T. A. DeLong, and J. Profeta, "An Algorithm Based Fault Tolerant Technique for Safety Critical Applications," *Reliability and Maintainability Symposium*, Jan. 1997.

[20] D. T. Smith, B. W. Johnson, and J. Profeta, "System Dependability Evaluation Via a Malicious Fault List Generation Algorithm", *IEEE Transactions on Computers*, (accepted for publication).

[21] J.L. Peterson, *Petri Net Theory and the Modeling of Systems*, Prentice-Hall, Englewood Cliffs, NJ, USA 1981.

[22] M. Ajmone Marsan, G. Conte and G. Balbo, "A Class of Generalized Stochastic Petri Nets for the Performance Evaluation of Multiprocessor Systems," *ACM Transactions on Computer Systems*, Vol.2, May 1984, pp. 93-122.

[23] W. N. Toy, "Fault-Tolerant Design of Local ESS Processor," *IEEE Proceedings*, Vol. 66, No. 10, October 1978, pp. 1126-1145.

[24] A. Law and W. Kelton, *Simulation Modeling and Analysis*, McGraw-Hill, Inc., New York,1991.

[25] J. Hammersley and D. C. Handscomb, *Monte Carlo Methods*, Methuen & Co. Ltd., London, 1964

[26] T. A. Delong, "Performance and Safety Analysis of a Microprocessor-based Embedded Control System using VHDL", Master of Science Thesis, University of Virginia, 1993.

[27] T. A. DeLong, B. W. Johnson, J. A. Profeta, III, and D. Bozzolo, "A Novel Fault Injection Technique for Behavioral-Level Modeling using VHDL," Proceedings Fall VHDL International Users Forum (VIUF), November 1994.

APPLICATIONS OF MASTER CURVE TECHNOLOGY

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1.0 Background

In the early 1970s, ASME adopted an approach to estimate the lower-bound fracture toughness values used to assess the integrity of pressure vessels, including nuclear reactor pressure vessels (RPVs). The ASME approach involves establishing an index temperature, RT_{NDT}, which positions both the static (K_{IC}) and dynamic (K_{IR}) fracture toughness transition curves on the temperature axis. RT_{NDT} is determined as prescribed by ASME NB-2331 [1] using a combination of Charpy V-notch and *NDT* testing conducted as per ASTM E208 and ASTM E23, respectively [2,3]. Also, nuclear RPV surveillance programs currently use the upward shift of the 30 ft-lb CVN transition temperature produced by irradiation to estimate the effect of irradiation on RT_{NDT} (see ASTM Standard E185 and USNRC Regulatory Guide 1.99, Revision 2, respectively [4,5]). All of these approaches use correlations to fracture toughness, rather than direct measurements of fracture toughness, due to the extremely large specimen size required to measure "valid" linear-elastic fracture toughness values in accordance with ASTM E399 [6].

Developments since the early 1970s provide a technical basis for fundamental improvements to these correlative techniques. In 1980, Landes and Schaffer noticed a statistical "size" effect for specimens failing by transgranular cleavage [7]. They demonstrated that larger specimens fail at lower toughness values than smaller specimens, even when the severe size requirements of linear elastic fracture mechanics (ASTM E399) are satisfied. Beginning in 1984, Wallin and co-workers from VTT in Finland combined this "weakest link" size effect with micro-mechanical models of cleavage fracture [8-10]. Wallin proposed a model that accounts successfully for specimen size effects, and provides a means to calculate statistical confidence bounds on cleavage fracture toughness data. These concepts, combined with Wallin's observation (made first in 1984, and reinforced in 1991) that all ferritic steels exhibit the same variation of cleavage fracture toughness with temperature, gave birth to the notion of a "master" transition curve for all ferritic steels [8,11].

ASTM recently passed a standard (E1921) that describes how to measure the Master Curve index temperature, T_o, based on limited replicate testing [12]. E1921 also incorporates a modern understanding of elastic-*plastic* fracture mechanics, thereby permitting determination of T_o using specimens as small as a precracked Charpy. The potential for characterizing the entire fracture mode transition based on direct fracture toughness measurements using specimens already in surveillance programs has sparked considerable interest in Master Curve technology within the commercial nuclear power community. Recently, ASME published Code Case N-629 that permits use of a Master Curve-based index temperature (RT_{To}=T_o+35°F) as an alternative to traditional RT_{NDT}-based methods of positioning the ASME K_{IC} and K_{IR} curves [12]. This approach was adopted in ASME Code Case N-629 to enable use of Master Curve technology without requiring wholesale changes to the structure of the ASME Code.

Adoption of consensus standards that use the Master Curve signal its technological maturation. Indeed, some commercial nuclear power plants are interested in using Master Curve technology to support both licensing and plant life extension activities. Nevertheless, certain issues regarding the technical basis of the Master Curve, its breadth of applicability, and the prospects for using the Master Curve to assess the integrity of a nuclear RPV in a manner that does not degrade historically accepted margins of safety await final resolution. In view of this situation, the purpose of this paper is fourfold:

1. To describe briefly the Master Curve,
2. To enumerate what issues regarding the Master Curve require resolution to enable RPV applications,
3. To discuss on-going research projects concerning these topics, and assess how well these investigations address these issues
4. To outline one potential framework for nuclear RPV assessment using Master Curve technology.

2.0 The Master Curve

The Master Curve rests on the following two premises:

1. One Curve Shape: All ferritic steels exhibit the same variation of fracture toughness with temperature in the fracture mode transition between lower shelf and upper shelf.
2. Weakest Link Fracture: The failure of ferritic steels in fracture mode transition can be described by a weakest link process (i.e. the load sufficient to initiate a crack in a small material element in front of a pre-existing defect is also sufficient to completely fail the component).

Information available that supports these premises is as follows:

1. One Curve Shape: In their 1984 paper, Wallin, Saario, and Törrönen [8] allude to a possible physical rationale for a "master" curve when they propose that the temperature dependence of the plastic work to fracture in Griffith's local failure criteria [14] follows an exponential form that is common to many steels. This line of reasoning was abandoned, however, and by 1993 only an empirical basis for a single curve shape was discussed [11]. Indeed, NUREG/CR-5504, a document that provides the technical basis for ASTM E1921-97, mentions only an empirical basis [15]. Independent empirical assessments of a single curve shape invariably validate this finding, and show its validity for RPV steels both before and after irradiation [16-17].
2. Weakest Link Fracture: The assumption that a weakest-link mechanism controls cleavage fracture permits the derivation of simple mathematical formulae that describe the scatter of toughness data about the median, as well as the effect of sampling volume (i.e. crack front length) on observed values of fracture toughness [9 and 10, respectively]. Wallin demonstrated the consistency of these equations with experimental data, a process recently repeated for an expanded database of both irradiated and un-irradiated RPV steels [17].

The co-existence of a single median curve shape and a pre-defined dispersion of toughness about this median suggest that toughness tests need only measure the value of the median at a single temperature to define the variation of toughness, and statistical confidence bounds on toughness, at all temperatures. Since central tendency values (e.g. medians) can be defined accurately based on small sample sizes, only limited experimental replication is necessary. Moreover, the ability to scale toughness values as a simple function of crack front length provides the possibility of testing the small samples from surveillance capsules and scaling these toughness values to reflect the performance of a longer crack in a structure.

3.0 Master Curve Issues Relevant to Nuclear RPV Integrity Assessment

In 1997, NRC staff detailed a number of technical issues relevant to use of the Master Curve in the integrity assessment of nuclear RPVs in both a NRR User's Request [18], and in a paper published at the 1997 SMIRT conference [19]. The following list provides a condensation, further explanation, and refinement of these issues enabled by the significant research activity and industrial interest that has focused on this topic since 1997.

1. As would be true for any empirical approach, it is difficult to assess *a priori* applicability limits for the Master Curve. Specifically, does the Master Curve apply to all of the product forms and material strength grades used in the fabrication of nuclear RPVs? Furthermore, is the shape of the Master Curve stable with respect to irradiation embrittlement, or is it changed by irradiation as has been observed for Charpy V-notch testing?
2. ASTM Standard E1921-97 permits testing of fatigue pre-cracked specimens as small as 10mm square (i.e. precracked Charpy specimens). However, some evidence exists that T_o values estimated using such small specimens may be biased low due to constraint loss [20]. Is this bias adequately addressed by E1921-97?
3. ASME Code Case N-629 proposes that an alternative to RT_{NDT}, termed RT_{To}, be estimated by adding 35°F to the T_o value determined by E1921-97 testing. Does this value of 35°F provide K_{IC} and K_{IR} curves that bound fracture toughness data in the same manner that RT_{NDT} indexed K_{IC} and K_{IR} curves do? Additionally, RT_{NDT} is determined by testing specimens at impact loading rates while RT_{To} is determined by testing specimens at quasi-static loading rates. Should any consideration be given of loading rate effects on fracture toughness when defining RT_{To}, particularly if it is used to define the K_{IR} curve, a curve traditionally associated with crack arrest, not crack initiation.
4. In current regulations, the upward shift of the 30 ft-lb CVN transition temperature (CV_{30}) produced by irradiation is used to estimate the effect of irradiation on RT_{NDT}, and, consequently, on the K_{IC} and K_{IR} curves. Is there a relation between the shift in the 30 ft-lb CVN transition temperature produced by irradiation and the shift in RT_{To} produced by irradiation?
5. The current interest in using the Master Curve for nuclear RPV assessment appears motivated by the fact that RT_{To} usually positions the K_{IC} and K_{IR} curves at a lower temperature, thereby placing them closer to the fracture toughness data they are intended to represent, than would RT_{NDT}. While it is generally accepted that current vessel integrity assessment methodologies are conservative, a question arises regarding to what degree this implicit margin (the distance between the data and the Code curve) can be reduced without unfavorably impacting public safety. A factor that compounds developing a simple answer to this question is the highly variable nature of material properties, including fracture toughness, revealed in recent laboratory studies of RPV steels [21]. One study reported a variation of CV_{30} of up to 100°F in a single vessel weld. The presence of such uncertainty calls into question if any reduction of implicit margin is justified, and, if so, how this material variability can be accommodated in the Master Curve approach.

4.0 On-Going Research

Both the Electric Power Research Institute (EPRI) and the NRC Office of Nuclear Regulatory Research have on-going research programs focused on resolution of these issues. In this section we summarize the scope of these programs, highlight recent results, and point out areas in which progress is still lacking.

4.1 *NRC Issue 1. Master Curve Shape: Breadth of Applicability and Irradiation Effects*

Both EPRI and the NRC currently sponsor projects on this topic. The EPRI effort, being conducted by Professor Marjorie Natishan of the University of Maryland, focuses on determining the fundamental physical basis for the universal Master Curve shape [22-24]. Research conducted in this project demonstrates that the temperature dependency of cleavage fracture toughness depends *only* on the short-range barriers to dislocation motion established by the lattice structure. Other factors that vary with steel composition, heat treatment, and irradiation (e.g. grain size/boundaries, inclusions, precipitates, dislocation substructures) provide long-range barriers to dislocation motion. These factors influence the position of the transition curve on the temperature axis by changing the absolute value of toughness (i.e. T_o as determined by E1921-97). They do not, however, influence the temperature dependency of fracture toughness (i.e. the Master Curve shape). This understanding suggests that the myriad of metallurgical factors that influence absolute strength and toughness values exert little, if any, control on the form of the variation of toughness with temperature in fracture mode transition. Moreover, this understanding provides a theoretical basis that establishes, *a priori*, those conditions to which the Master Curve applies, and those to which it does not. Of particular interest in nuclear applications, these findings suggest that the Master Curve should apply to all RPV product forms over the strength range of practical interest. Furthermore, this work suggests that no embrittlement mechanism known to be operative in RPV steels is expected to change the Master Curve shape during the original design, or anticipated extended lifetime, of domestic PWRs.

NRC research on the shape of the Master Curve is being conducted by Mikhail Sokolov and Randy Nanstad of the Oak Ridge National Laboratory (ORNL), and by Professor Robert Odette of the University of California at Santa Barbara. In 1996 Sokolov and Nanstad assessed the effect of irradiation on Master Curve shape using fracture toughness data from laboratory irradiation studies [16]. This investigation demonstrated that irradiation does not alter the Master Curve shape. There was some indication of a slight shape change at high irradiation levels, but the finding was not judged to be statistically significant. This indication of the potential for Master Curve shape change at high fluence, combined with concern that a synergistic thermal / irradiation embrittlement mechanism could promote inter-granular failure, led to initiation of a project, currently underway, to assess experimentally the Master Curve shape for a RPV steel exhibiting high sensitivity to neutron irradiation. Specimens have recently been removed from a test reactor, and will be tested at ORNL in the years 2000-2001.

A project being conducted under NRC sponsorship by Professor Robert Odette at the University of California (Santa Barbara) is similar in intent to investigations conducted at the University of Maryland: ascertain a physical basis for a universal Master Curve shape. This project is just beginning; consequently no results are available at this time.

4.2 *NRC Issue 2. Constraint Loss Bias in T_o Values Determined from Small Specimens*

ASTM standard E1921-97 includes the following deformation limit on fracture toughness values that are considered "valid":

$$MIN(b,a,B) \geq \frac{30 \cdot K_{Jc}^2}{\sigma_{ys} \cdot E} \tag{1}$$

Eq. (1) was adopted as a compromise between experimental evidence that suggested a less restrictive limit (i.e. below 30) might be possible, and the results of finite element analyses that pointed to the need for more restrictive limits (i.e. above 30). Subsequent NRC sponsored research performed by Professor Robert Dodds of the University of Illinois [20] predicts the magnitude of the bias introduced to T_o by data sets containing specimens, particularly precracked Charpy specimens, that challenge the eq. (1) limit. Experimental assessments of this bias, published by both NRC and EPRI contractors [25-26], show a bias having the functional form predicted by Dodds, but having a lower magnitude, a magnitude that is sometimes difficult to distinguish from zero. Work on this topic continues to be conducted by both NRC and EPRI contractors (Mikhail Sokolov of ORNL, Robert Tregoning of the Naval Surface Warfare Center, Professor James Joyce of the U.S. Naval Acadamy, and William Server of ATI Consulting).

4.3 NRC Issue 3. Relationship Between RT_{NDT} and RT_{To}

The PVRC Task Group on Master Curve Applications developed a phased strategy for incorporation of Master Curve concepts into the ASME Code. In the short term this strategy advocates developing a Master Curve-based index temperature for the ASME K_{IC} and K_{IR} curves to use as an alternative to RT_{NDT}. In the long term the group envisions eliminating the K_{IC} and K_{IR} curves and, instead, employing a methodology that uses the Master Curve directly.

While work toward the long-term goal is only just beginning, the short-term goal was accomplished with the adoption of Code Cases N-629 by ASME Section XI, a committee that includes voting representatives from both industry and from the NRC. This Code Case provides the following formula for RT_{To}, the Master Curve-based index temperature:

$$RT_{To} \equiv T_o + 35°\mathrm{F} \tag{2}$$

An industry group working under a combination of EPRI and Owners' Group funding developed the technical basis for Code Case N-629 [27]. This document points out that even though there is no direct correlation between RT_{NDT} and RT_{To} [28], an approach that ensures that RT_{To} continues to perform the same function as RT_{NDT} can relate the two quantities. Historically, a K_{IC} curve indexed to RT_{NDT} has provided an appropriate lower bound to available fracture toughness data. The challenge in selecting the numeric value of 35°F in eq. (2) therefore involved agreeing on and quantifying the degree of margin implicit to a lower bound K_{IC} curve indexed to RT_{NDT}. As illustrated in Figure 1 using data for RPV steels [29], the level of implicit margin (i.e. the horizontal distance between fracture toughness data and a RT_{NDT}-indexed K_{IC} Curve) varies considerably. Because of this, the committee decided that in the past one could only rely on the level of margin implicit to the most limiting material (i.e. the material having fracture toughness data closest to the bounding curve). Maintaining the margin historically associated with HSST Plate 02 (+17°F on Fig. 1) therefore became the mark for RT_{To} to achieve (the negative implicit margin on Fig. 1 for a weld was not available to the committee that drafted reference [27]). The committee determined that the 35°F margin in eq. (2) exceeds that implicit to HSST-02 by 18°F. Consequently, RT_{To} values determined by eq. (2) include implicit margins that equal or exceed those that have been accepted historically.

Work on establishing bounding K_{IC} curves using RT_{To} has also been conducted under NRC sponsorship at ORNL [30]. This work expressed the view the K_{IC} curve, which when drawn in 1973 provided an absolute bound all to data in fracture mode transition (i.e., at all temperatures exceeding T- RT_{NDT} = -100°F), should continue to exhibit this bounding characteristic. Based on this criterion, the numeric value in eq. (2) would be 61°F. One difficulty with this definition of RT_{To} is the implication that the regulatory definition of the required implicit margin changes as more data becomes available.

As described here, even though the Code Case N-629 and has been adopted, some disagreement remains regarding the appropriate numeric value in eq. (2). The purpose of this paper is not to resolve this issue, but merely to point out that disagreement exists. Furthermore, it does not appear that additional technical information is needed to resolve the issue, but rather a broader and more consistent understanding of what the requirements of existing ASME code and existing regulations actually are.

4.4 NRC Issue 3 . Loading Rate Effects

Because failure of a RPV would likely involve a rapidly propagating crack, the fracture integrity assessment methodology used for RPVs must account for the effect of loading rate on fracture toughness. However, the way the loading rate question was posed previously [18-19] is somewhat confusing because rate effects do *not* enter the methodology via the loading rate used to determine the index temperature. Rather, rate effects enter the methodology via the separation between the static (K_{IC}) and dynamic (K_{IR}) fracture toughness transition curves. This separation is currently fixed at 53°F (at $K_{IC} \approx 50$ ksi$\sqrt{\text{in}}$) irrespective of loading rate differential, strength level, and irradiation level. Nevertheless, available empirical evidence demonstrates that increases in material strength mitigate the effect of loading rate on fracture toughness in the transition regime [31-32]. Consequently, the current use of a fixed separation between static and dynamic transition curves, while conservative, is not only inaccurate, but also physically inappropriate.

In summary, rate effects enter a RPV integrity assessment methodology not via the loading rate used for tests that determine the index temperature, but rather through the separation between the quasi-static and high-rate fracture toughness transition curves. Current technology uses high loading rate tests to establish the temperature RT_{NDT}, which is used to index *both* the K_{IC} and the K_{IR} curves. Equally, it should be possible to construct a methodology using an index temperature determined under quasi-static loading rates (e.g. RT_{To}). Recently completed and on-going research pertinent to loading rate effects on fracture conducted within the United States nuclear community includes the following:

- Research performed for the NRC by Professor James Joyce of the United States Naval Academy demonstrates that the Master Curve can be applied to the characterization of fracture mode transition at high loading rates [33]. Joyce provides data regarding the effect of loading rate on T_o for ASTM A515 Gr. 70 (a non-nuclear grade of pressure vessel steel).
- In a follow-on to the Joyce study, Link and Graham quantified the effect of loading rate on T_o for the nuclear pressure vessel steel ASTM A533B [34].
- In research sponsored by the B&W Owners' Group, Ken Yoon has cited these experimental studies as evidence that maintaining the current fixed separation of 53°F between K_{IC} and K_{IR} curves indexed to RT_{To} is appropriate [35]. This finding is based on a maximum loading rate during PTS of 10 MPa$\sqrt{\text{m}}$/sec.
- On-going research performed for the NRC by Robert Tregoning of the United States Navy and by Professor Robert Dodds of the University of Illinois is aimed at calibrating a Weibull failure model to describe the combined effects of loading rate and temperature on the fracture toughness reactor pressure vessel steels. Weibull models of this type have previously been successful at correlating constraint and bi-axiality effects on toughness at a single temperature in fracture mode transition [36]. While this approach is expected to be successful in quantifying the functional form for rate effects, its generic application may be hampered because, to date, the material coefficients in the Weibull model lack the physical basis that would provide insight regarding the model's applicability to all conditions of interest.

Finally, full resolution of this issue requires consideration of crack arrest. In current RPV integrity assessment methodologies, the K_{IR} curve represents the lower bound of *both* dynamic initiation and crack arrest data. Also, in a PTS calculation, a vessel is not considered to have "failed" unless an initiated crack cannot be arrested [37]. Treatment of crack arrest within the current technical understanding of the Master Curve does not appear possible, for while crack arrest is a transition fracture phenomena, it clearly cannot follow a weakest link model for crack initiation. To date the crack arrest transition temperature has only been correlated to T_o, rather than predicted from it [38]. However, it may not be necessary to provide a technical resolution of issues related to crack arrest. Defining vessel "failure" as a crack that has initiated and cannot be arrested is an engineering decision, and indeed not the most conservative failure criteria. For example, in France initiation alone is all that is required to produce vessel "failure," thereby entirely avoiding the questions and uncertainties surrounding crack arrest measurement and calculation.

4.5 *Crack Front Length Effects (Issue Not Identified Previously)*

The Master Curve incorporates the following relationship between fracture toughness and the length of the crack front:

$$K_{Jc}\Big|_{Thick2} = 20 + \left(K_{Jc}\Big|_{Thick1} - 20\right)\cdot\left[\frac{B_{Thick1}}{B_{Thick2}}\right]^{1/4} \qquad (3)$$

where the units on K_{Jc} are MPa$\sqrt{}$m. Eq. (3), which is based on a weakest-link model for fracture by transgranular cleavage, predicts a decline in fracture toughness with increasing crack front length, a prediction in accord with considerable experimental evidence for fracture test specimens [17]. Equation (3) represents a significant departure from the current use of fracture mechanics in the ASME code. Currently, toughness and crack fromt length are treated as independent variables, yet eq. (3) links them. This change produces challenges for adoption of the Master Curve into ASME Code for the PVRC long-term goal, but not for the PVRC short-term goal (see Section 4.3), as described below:

- PVRC Short Term Goal - RT_{To} Indexed K_{IC} and K_{IR} Curves: This methodology uses the Master Curve to obtain an index temperature (RT_{To}) while maintaining the notion that bounding curves of a large experimental database (K_{IC} and K_{IR}) can be positioned using this index temperature. This approach makes the fracture toughness / crack length linkage implicit to and specified by the size of specimens in the experimental database used to establish the bounding curves. Thus, the old (RT_{NDT} indexed curves) and the new (RT_{To} indexed curves) methodologies become equivalent provided the bounding curves for both are developed from the same experimental database.
- PVRC Long Term Goal - Replacement of K_{IC} and K_{IR} curves with the Master Curve: Here the linkage of crack front length and toughness represented by eq. (3) is an inescapable component of the methodology because Master Curve provides a means to calculate the the bounding curve (i.e., a xx% tolerance bound). To enable application of the Master Curve to structures, a form of eq. (3) that addresses non-straight fronted fatigue cracks, and can differentiate between statistical size effects and constraint effects is needed. Four on-going research projects address these goals.
  1. The NRC has sponsored an experimental program with Professor Robert Odette at the University of California (Santa Barbara). The aim of this program is to separate and quantify the competing "size" effects (i.e., constraint *vs.* statistical) by conducting a controlled set of fracture experiments using a RPV steel. Preliminary results demonstrate that the form of eq. (3) is appropriate for RPV steels [39].

2. The NRC has sponsored research concerning Weibull models for failure in fracture mode transition.
   - Professor Robert Dodds of the University of Illinois and Robert Tregoning of the United States Navy work jointly on a project concerning fracture initiated from semi-elliptical surface cracks. These models have proved successful at predicting the combined effects of constraint loss and statistical size on semi-elliptical surface cracks loaded in a combination of bending and tension [40].
   - A related NRC sponsored research project conducted by Richard Bass of ORNL focuses on applying/modifying the Illinois Weibull failure models for surface cracks subjected to bi-axial loading [36]; follow-on work on buried cracks will be conducted in the year 2000 and beyond.

   These efforts will provide predictive tools for non-straight fronted cracks that can differentiate between constraint loss and statistical size effects. However, an uncertainty concerning the appropriate stress parameter to use in the Weibull Stress calculation remains. Dodds has reported success using the maximum principal stress for uniaxially loaded samples [40], while Bass finds use of the hydrostatic stress necessary to correlate bi-axial effects [36]. Heretofore, the stress parameter used in the Weibull Stress calculation has been determined empirically as a back-fit to experimental data, making it impossible to use this methodology in a predictive manner. Some resolution of this discrepancy, and a metallurgical justification of the appropriate stress to use in the Weibull calculation is therefore needed before these methodologies can be applied with confidence to all conditions of interest.
3. Finally, EPRI has sponsored a task with Randy Lott of Westinghouse and William Server of ATI Consulting to establish how the size of postulated cracks (surface and embedded) in vessels should be represented to the Master Curve to support the PVRC long term goal of replacing the K_{Ic} and K_{IR} curves with the Master Curve.

Outside of flaw specific-assessments performed according to ASME Section XI (IWB-3500, IWB-3600), non-straight fronted cracks are currently considered in reactor pressure vessel integrity analysis in two areas: in the plant-specific PTS analysis described in 10CFR50.61 and performed in accordance with Regulatory Guide 1.154, and in the calculation of permissible limits on heat-up and cool-down performed in accordance with ASME Section XI Appendix G. The flaws considered in PTS calculations are of a realistic dimension because they are intended to represent the flaws in actual RPVs. Thus, resolution of the technical issue on toughness scaling expressed previously should resolve the crack front length and constraint issues for PTS. Conversely, heat-up and cool-down curves are calculated using a flaw that penetrates one-quarter of the way through the reactor pressure vessel wall and has a 6:1 ratio of surface breaking length to depth. This flaw is much larger than expected or observed in any operating RPV, making the flaw size an apparent conservatism implicit to this analysis methodology. In an 8-in. thick vessel this flaw would have a crack front length of 14-in. Thus, before the Master Curve can replace completely the K_{IC} and K_{IR} curves for Appendix G analyses, a reconciliation of the ¼-T flaw methodology and the Master Curve approach is needed.

4.6 <u>NRC Issue 4</u>. Relationship Between Irradiation Induced Shifts in CV_{30} and RT_{To}

RPV integrity analysis requires an estimate of the fracture toughness after irradiation. Determination of RT_{NDT} for the unirradiated steel (the *IRT*, or initial reference temperature) is the first step in this evaluation. The *IRT* is modified by adding (a) the shift in Charpy transition temperature at 30 ft-lbs produced by irradiation (ΔRT_{NDT}), and (b) a margin term (M) to account for uncertainties [5]. This calculation produces a reference temperature adjusted for the effects of irradiation, or *ART*. Reg. Guide 1.99 (Rev. 2) defines the *ART* as follows:

$$ART = IRT + \Delta RT_{NDT} + M \qquad (4)$$

where

$$\Delta RT_{NDT} = (CF) f^{(0.28-0.1\log f)}$$

$CF =$ the Chemistry Factor, either from a fit to Charpy surveillance data, or estimated from copper content, nickel content, and product form

$f =$ fluence / 10^{19} n/cm^2

$$M = 2\sqrt{\sigma_I^2 + \sigma_\Delta^2}$$

$\sigma_I =$ the uncertainty in IRT

$\sigma_\Delta =$ the uncertainty in ΔRT_{NDT}.

The functional form of the fluence function used in determining the irradiation shift, $f^{(0.28-0.1\log f)}$, was established by curve-fitting a database of 177 30 ft-lb Charpy shift values [41]. A question therefore arises regarding the applicability of eq. (5) to prediction of irradiation shifts in fracture toughness (i.e. T_o) data. However, irradiation shifts in both Charpy and fracture toughness transitions are largely controlled by increases of material flow strength produced by irradiation. It is therefore reasonable to assume that a fluence function that describes shifts of Charpy transition temperature might also model shifts in the fracture toughness transition temperature well. Work sponsored by both the NRC and by EPRI has investigated this possibility. Under NRC sponsorship, Mikhail Sokolov and Randy Nanstad of ORNL compared irradiation shifts of both CVN energy and fracture toughness transition [16]. This comparison showed a 1:1 correlation for welds (42 data points). Conversely, examination of 47 plate materials shows that irradiation shifts the fracture toughness transition temperature 16% more than it does the CVN transition temperature. In both cases the relationship between the two transition temperatures was linear. In more recent EPRI sponsored work, Mark Kirk and Randy Lott of Westinghouse and William Server of ATI consulting compare available data on T_o shifts produced by irradiation to the functional form of eq. (4), which was established for CVNs. These investigators conclude that, based on available empirical evidence for 16 RPV steels, the Reg. Guide 1.99 (Rev. 2) fluence function provides a reasonable description of the shift in T_o produced by irradiation [42].

While both the NRC and EPRI sponsored work provides encouraging results, the high cost of irradiated material testing will likely preclude development of sufficiently well populated database of T_o shift values to test empirically the appropriateness of eq. (4) for all conditions of interest. Consequently, resolution of this issue seems to rest with establishing a sound basis for *why* the T_o and CVN shifts *should* be the same. Existence of such a rationale, which is not currently being investigated by either the NRC or EPRI, would pave the way for establishing the appropriate functional form for T_o shifts based on NRC research being conducted by Ernie Eason of M&CS Inc. Eason is using in excess of 700 measured CVN shift values to both improve the statistical robustness of eq. (4), and to reflect the current best understanding of the physical mechanisms responsible for irradiation damage [43].

4.7 NRC Issue 5. Margin Adequacy and Material Variability

Interest in application of Master Curve concepts to nuclear RPV integrity assessments has arisen in the past few years due to the potential for obtaining a more accurate assessment of the degree of irradiation embrittlement in an operating RPV. This improved accuracy relative to current techniques occurs because a direct measurement is made of the quantity of interest, and because of the better physical basis for the Master Curve *vs.* the empirical / correlative basis for a K_{IC} curve indexed to RT_{NDT}. Figure 2 illustrates that this accuracy improvement results in consistent positioning of a bounding K_{IC} curve relative to fracture toughness data, a consistency that cannot be achieved using current RT_{NDT}-based techniques. However, as also illustrated by Fig. 2, this accuracy improvement usually reduces the degree of conservatism implicit to current procedures. Figure 3 demonstrates that approximately 80% of the time the Master Curve index temperature RT_{To} places a bounding curve closer to the actual fracture toughness data than would RT_{NDT}.

Figures 1 through 3 illustrate that a widely varying degree of implicit margin exists in current assessment methodologies. This variation complicates development a simple answer to the question "How much margin is enough?" Further complications include the fact that the implicit margin on toughness needed to achieve the SECY-82-465 limit on through wall cracking of 5×10^{-6} events per reactor year has never been specified for existing (RT_{NDT}-based) methods. One final complicating factor arises in the highly variable nature of fracture toughness and CVN properties, even within a single weld. In work sponsored by the NRC, Don McCabe of ORNL reported a variation of CV_{30} of up to 100°F in a single vessel weld [21]. The presence of such uncertainty calls into question if any reduction of implicit margin is justified, and, if so, how this material variability can be accommodated in the Master Curve approach. Pilot studies on how material uncertainties have been addressed in the past, and on the development of frameworks to address this issue for the Master Curve are currently sponsored by both EPRI (at Westinghouse), and by the NRC (at ORNL). In recent presentations, Randy Lott of Westinghouse has proposed a framework to classify the different sources of material uncertainty based on the size scale over which the material exhibits heterogeneous behavior [44]. This framework, illustrated in Figure 4, builds on the concept of a material class expressed within both Regulatory Guide 1.99 Revision 2 and 10CFR50.61.

Both the EPRI and NRC sponsored studies are expected to provide important background and contextual information. Nevertheless, since overall plant risk is influenced by a myriad of factors of which fracture toughness is but one, questions concerning required margins for fracture toughness cannot be addressed in isolation. One potential integrated approach that considers multiple factors simultaneously is discussed in Section 5.0.

5.0 An Integrated Assessment of Master Curve Applications in Nuclear RPV Integrity Assessment: One Potential Framework

Here we propose one potential integrated framework for assessing the suitability of Master Curve-based assessment of RPV integrity assessment. This proposal contains two components:

1. A gated approach to address uncertainties associated with the material, and with the state of knowledge about the material.
2. A probabilistic risk assessment calculation.

USNRC Regulatory Guide 1.99 Revision 2 provides current guidance regarding how to estimate the effect of neutron irradiation embrittlement on the fracture toughness of a nuclear RPV. This effect is quantified using an Adjusted Reference Temperature, or *ART*. *ART* is a value of RT_{NDT} *adjusted* to account for the effects of irradiation. Reg. Guide 1.99 (Rev. 2) is "gated" in the sense that multiple paths are available to estimate *ART*, the path employed depending on the state of knowledge of the material. Currently, if measurements of transition temperature quantities (i.e. T_{NDT} and CV_{30}, and ΔCV_{30}) are available through initial conformance testing or through surveillance programs, these values provide the basis for estimating *ART*. If this information is not available, generic unirradiated values are assigned based on material class, and an irradiation shift is estimated from the chemical composition of the material. Margins for knowledge and material uncertainties are applied to all of these estimates, the magnitude of this margin distinguishing between the different paths. Smaller margins are required when measurements of transition temperature quantities are available, while larger margins are applied if less is known about the material (e.g. only generic material class and chemistry).

Figure 5 illustrates a framework of this type, expanded to admit the possibility of using fracture toughness measurements to estimate T_o, as well as using CVN measurements to estimate CV_{30}[1]. This framework incorporates a margins strategy to address material variability that builds on the concept of a generic material class expressed within both Regulatory Guide 1.99 Revision 2 and 10CFR50.61. This diagram also illustrates that the results of any framework of this type will provide input to a probabilistic risk assessment (PRA) code (e.g. the FAVOR code as developed for the NRC by ORNL [37]). The PRA is used to assess the impact of the proposed framework for fracture toughness on achieving the SECY-82-465 criteria that the probability of through wall cracking should not exceed $5x10^{-6}$ per operating year [43]. The PRA calculations consider all of the relevant influences on reactor vessel integrity (including fracture toughness) simultaneously. Consequently, the PRA calculations provide a mechanism to assess the influence of using the Master Curve in a fully integrated fashion.

6.0 Developments Needed to Enable Use of the Master Curve

Successful implementation of the framework illustrated in Fig. 4 depends on continued technical progress in the coming years. Assumed outcomes for current research initiatives are presented here, along with an assessment of the work necessary to realize them. Both the assumed outcomes and the needed additional work, if any, are shown **in boldface type**. Outcomes are summarized in three areas: (1) research, (2) codes and standards, and (3) methodology.

6.1 Research Outcomes

1. **The Master Curve retains its universal shape for all material and irradiation conditions of interest through the license extension period (i.e. through 60 EFPY).** Current research sponsored by both EPRI and the NRC appears appropriate to validate this conclusion. Successful resolution of this issue is anticipated through completion of currently planned research activities.

2. **The effect of loading rate on the Master Curve transition temperature will be quantified such that the T_o for a high rate Master Curve can be reliably calculated based on knowledge of the material strength, of the quasi-static T_o (determined as per E1921), and of the loading rate of interest. Routine dynamic testing to determine a high rate T_o directly will not be necessary.** Considerable favorable empirical evidence exists suggesting that this result is possible. Additionally, on-going NRC sponsored research should provide the methodologies needed to predict dynamic T_o values from static T_o values. However, the bounds of applicability of these methodologies remain to be demonstrated, and a comprehensive experimental demonstration appears economically unfeasible. Nevertheless, **some demonstration must be made that these techniques can be expected to apply to all material conditions of interest through the license extension period.**

3. **A size effect model that differentiates between the combined constraint and statistical effects on toughness for non-straight fatigue cracks will be developed.** Current research sponsored by the NRC and EPRI appears adequate to achieve this result. However, an

[1] Favorable resolution of a number of the technical issues detailed in Section 4.0 was assumed in making this diagram. These assumptions are discussed in Section 6.0.

uncertainty remains regarding the input stress parameter. Different stress parameters, which have been selected empirically, are currently needed to accurately predict failure for different loading conditions, suggesting that the models have yet to fully capture the physics of the underlying fracture process. **Research is needed to provide a metallurgical justification for the appropriate stress input before these models can be applied with confidence to all conditions of interest.**

4. **Research is needed to provide a physical rationale for why shifts in T_o and CVN caused by irradiation should be the same.** Were this result available, it would pave the way for establishing the appropriate functional form for T_o shifts based on NRC research currently being conducted by Ernie Eason of M&CS Inc. While neither the NRC or EPRI are currently pursuing such work, available empirical evidence suggests the likelihood of a favorable outcome. Pursuit of a purely empirical solution is not envisioned given the high cost of irradiated testing.

5. **Research is needed to establish the margins that should be applied to ART estimates to address material and state of knowledge uncertainties.** One possible solution would be a gated approach in which progressively smaller margins are applied to measured values as the samples used to determine the measured values become progressively more "like" the operating reactor. Such a gated approach would maintain the framework currently employed within Regulatory Guide 1.99 Revision 2 and 10CFR50.61. Pilot studies on this topic are currently sponsored by both EPRI (at Westinghouse), and by the NRC (at ORNL). These studies are expected to provide important background and contextual information, however neither study is currently has the resources needed resolving the issues surrounding material variability and margins on a technical basis. While the numerical values for these margins will unquestionably be developed based on data, it also seems important to understand the reasons for differences in the data within a metallurgical context to help guard against the wholesale application of worst-case margins in situations where such application is inappropriate. Failure to establish such margins, and their technical basis, makes application of Master Curve technology problematic. **Research is therefore needed to provide a physically realistic, quantitative assessment of the margins that should be used when Master Curve technology is applied to the fracture integrity assessment of nuclear RPVs.**

6.2 *Codes and Standards Outcomes*

1. **A consensus will emerge from on-going EPRI and NRC research regarding the potential for constraint loss when testing precracked CVN specimens removed from surveillance capsules to determine T_o using E1921.** This consensus will either validate the current constraint limit (eq. (1)), or will provide a means to correct for the bias introduced by constraint loss. In either event, the possibility of using the precracked Charpy specimen within ASTM Standard E1921 will remain. The technical information needed to reach this consensus appears to be currently available.

2. **A consensus will emerge from on-going NRC / industry interactions regarding what the appropriate relationship is between RT_{To} and T_o (eq. (2)).** It does not currently appear that additional technical information is needed for resolution of this issue, but rather a broader and more consistent understanding regarding what the requirements of existing ASME code and existing regulations actually are. One appropriate forum in which this understanding can emerge is within the PVRT Task Group on the Master Curve. This issue is a small component of broader questions regarding the determination of appropriate margins (see Research Outcome #5).

3. **The procedure in ASME Appendix G that requires assumption of a 6:1 ¼-T flaw for heat-up and cool-down calculations will need to be reconciled with the Master Curve Approach.** It is not envisioned that any plausible outcome from research (see Section 6.1) will render such a large crack benign when scaled using a weakest-link model. The 6:1 ¼-T flaw was originally adopted in the Code as a conservatism in a time when no linkage between crack size and fracture toughness was anticipated. The linkage of these two variables imposed by the Master Curve necessitates modification of the 6:1 ¼-T requirement in the direction of increased realism.

6.3 *Methodology Outcomes*

1. **A means to address crack arrest in a PTS calculation will be established.** This issue could be resolved non-technically by adopting the conservative definition of vessel failure as crack initiation, in a manner consistent with the approach taken in France. Alternatively, the current definition of failure (an initiated crack that cannot be arrested) could be retained and a correlative procedure adopted to estimate an index temperature for crack arrest toughness [38]. Ideally this correlative procedure should be supported by a physical model to ensure the appropriateness of the correlation to all conditions of interest.

7.0 Summary

The Master Curve, as introduced by Wallin and co-workers in 1984, has evolved into a mature technology for characterizing the fracture toughness transition of ferritic steels. Considerable empirical evidence provides testament to the robustness of the Master Curve procedure. In 1997, NRC staff detailed several technical issues requiring resolution prior to acceptance of Master Curve technology for application to the fracture integrity assessment of nuclear RPVs. Current and recently completed research programs sponsored by both EPRI and the NRC focus on closure of these issues. This paper reviews the issues detailed in 1997, comments on their continued relevance in light of recent research results, and details areas where additional research may be warranted. Further research is suggested in two general areas: development of the theoretical justifications for empirical trends that will enable confident application of Master Curve technology to all conditions of interest, and development of a means to address material variability issues so that an adequate level of margin against vessel fracture continues to be maintained. A better resolution of these issues than appears possible using currently available information is viewed as a prudent step before widespread application of the Master Curve to nuclear RPVs is possible.

This paper also outlines a process for assessing how to incorporate the Master Curve into the existing regulatory framework. This assessment includes a proposed methodology to address material variability and margins, and the performance of probabilistic fracture mechanics calculations to assess the effect of this methodology on the overall risk of vessel failure. It is expected that this integrated assessment will rely principally on currently available data, while the research projects mentioned in the preceding paragraph will provide a theoretical justification for the appropriateness and safety of applying the Master Curve to all conditions of interest. Consequently, it should be possible for the remaining research and the integrated assessment of Master Curve applicability to nuclear RPV assessment to proceed in parallel.

8.0 References

1. ASME NB-2331, 1998 ASME Boiler and Pressure Vessel Code, Rules for Construction of Nuclear Power Plants, Division 1, Subsection NB, Class 1 Components

2. ASTM E208-95a, "Standard Test Method for Conducting Drop-Weight Test to Determine Nil-Ductility Transition Temperature of Ferritic Steels," ASTM 1998.
3. ASTM E23-96, "Standard Test Methods for Notched Bar Impact Testing of Metallic Materials," ASTM, 1998.
4. ASTM E185-94, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels," ASTM, 1998.
5. Regulatory Guide 1.99, Revision 2, *Radiation Embrittlement of Reactor Vessel Materials, U.S. Nuclear Regulatory Commission*, May 1988.
6. ASTM E399-90, "Standard Test Method for Plane-Strain Fracture Toughness Testing of Metallic Materials," ASTM, 1998.
7. Landes, J.D., and Shaffer, D.H., "Statistical Characterization of Fracture in the Transition Region," *Fracture Mechanics, 12th Conference, ASTM STP 700,* American Society for Testing and Materials, pp. 368-382, 1980.
8. Wallin, Saario, and Törrönen, "Statistical Model for Carbide Induced Brittle Fracture in Steel," *Metal Science,* 18, pp. 13-16, 1984.
9. Wallin, K., "The Scatter in K_{Ic} Results," *Engineering Fracture Mechanics*, **19**(6), pp. 1085-1093, 1984.
10. Wallin, K., "The Size Effect in K_{Ic} Results," *Engineering Fracture Mechanics*, **22**, pp. 149-163, 1985.
11. Wallin, K., "Irradiation Damage Effects on the Fracture Toughness Transition Curve Shape for Reactor Vessel Steels," In*t. J. Pres. Ves. & Piping*, **55**, pp. 61-79, 1993.
12. ASTM E1921-97, "Test Method for Determination of Reference Temperature, T_o, for Ferritic Steels in the Transition Range," ASTM, 1998.
13. ASME Code Case N-629, "Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials for Section XI".
14. Griffith, A., "The Phenomena of Rupture and Flow in Solids," *Philosophical Transactions, Series A,* Vol. 221, pp. 163-198, 1920.
15. Merkle, J.G., Wallin, K., McCabe, D.E., "Technical Basis for an ASTM Standard for Determining the Reference Temperature, T_o, for Ferritic Steels in the Transition Range," NUREG/CR-5504, 1998.
16. Sokolov, M.A., and Nanstad, R.K., "Comparison of Irradiation Induced Shifts of K_{Jc} and Charpy Impact Toughness for Reactor Pressure Vessel Steels," *ASTM STP-1325,* American Society of Testing and Materials, 1996.
17. Kirk, M., Lott, R., Kim, C., and Server, W., "Empirical Validation Of The Master Curve For Irradiated And Unirradiated Reactor Pressure Vessel Steels," Proceedings of the 1998 ASME/JSME Pressure Vessel and Piping Symposium, July 26-30, 1998, San Diego, California, USA.
18. NRR Memorandum to RES dated April 22, 1997, "Request for Research on Characterization of Reactor Pressure Vessel Material Toughness by the Master Curve Method."
19. Mayfield, M.E., et al., "Application of Revised Fracture Toughness Curves in Pressure Vessel Integrity Analysis," Paper #106, SMIRT Conference, Lyon, France, August, 1997.
20. Ruggieri, C., Dodds, R.H., and Wallin, K., "Constraint Effects on Reference Temperature, T_o, for Ferritic Steels in the Transition Region," *Engineering Fracture Mechanics*, 1998, Vol. 60, pp. 19-36.
21. McCabe, D.E., et al., "Unirradiated Material Properties of Midland Weld WF-70," NUREG/CR-6249, U.S. Nuclear Regulatory Commission, 1994.
22. Natishan, M. and Kirk, M., "A Micro-mechanical Evaluation of the Master Curve," *Fatigue and Fracture Mechanics, 30th Volume, ASTM STP-1360,* K. Jerina and P. Paris, Eds., American Society for Testing and Materials, 1998.

23. Natishan, M.E., Wagenhoefer, M., and Kirk, M.T., "Dislocation Mechanics Basis and Stress State Dependency of the Master Curve," *Fracture Mechanics, 31st Symposium, ASTM STP 1389*, K. Jerina and J. Gahallger, Eds., American Society for Testing and Materials, 1999.
24. Natishan, M. and Kirk, M., "A Physical Basis for the Master Curve," *Proc. of the 1999 ASME Pressure Vessel and Piping Conference*, ASME, July 1999.
25. Joyce, J., and Tregoning, R., "Development of the T_o Reference Temperature from Precracked Charpy Specimens," *Proc. of the 1999 ASME Pressure Vessel and Piping Conference*, ASME, July 1999.
26. Kirk, M., Lott, R., Server, W., Hardies, R., and Rosinski, S., "Bias and Precision of T_o Values Determined using ASTM Standard E1921-97 for Reactor Pressure Vessel Steels," *19th International Symposium on the Effects of Irradiation on Materials, ASTM STP-1366*, M.L. Hamilton, S. Rosinski, M. Grossbeck, and A. Kumar, Eds., American Society for Testing and Materials, 1998.
27. "Application of Master Curve Fracture Toughness Methodology for Ferritic Steels," EPRI-TR-108390, 1998.
28. Sokolov. M.A., "Statistical Analysis of the ASME K_{Ic} Database," *Journal of Pressure Vessel Technology*, Vol. 120, February 1998.
29. Master Curve Database, Electric Power Research Institute, August 1999.
30. Bass, B.R., McAfee, W.J., and Pennel, W.E., "Evaluation of RPV Structural Weld Fracture Toughness Data using ASTM Reference Temperature T_o," ORNL/NRC/LTR-98-08, 1998.
31. Barsom, J.M., and Rolfe, S.T., *Fracture and Fatigue Control in Structures, Applications of Fracture Mechanics*, Prentice Hall, 1987.
32. Wallin, K., "Loading Rate Effect on the Master Curve T_o," Paper IIW- X-1403-97, 1997.
33. Joyce, J.A., "On the Utilization of High-Rate Pre-Cracked Charpy Test Results and the Master Curve to Obtain Accurate Lower Bound Toughness Predictions in the Ductile-to-Brittle Transition," *Small Specimen Test Techniques, ASTM STP 1329*, Corwin, W., Rosinski, S., and VanWalle, E., Eds., American Society for Testing and Materials, 1997.
34. Link, R., and Graham, S., "Dynamic Fracture Toughness Initiation of A533 Grade B Steel Plate, NUREG CR-6512, United States Nuclear Regulatory Commission, 1997.
35. Yoon, K., VanDerSluys, A., and Hour, K., "Effect of Loading Rate on Fracture Toughness of Pressure Vessel Steels." *Proc. of the 1999 ASME Pressure Vessel and Piping Conference*, ASME, July 1999.
36. Bass, B.R., et al., "Research Results Supporting Enhanced Fracture Analysis Methods for Nuclear Reactor Pressure Vessels," NUREG/CR-6657 (In Review, November 1999).
37. Dickson, T.L., "An Overview of FAVOR: A Fracture Analysis Computer Code For Nuclear Reactor Pressure Vessels," Transactions of the 13th International Conference on Structural Mechanics in Reactor Technology (SMiRT 13), Vol. IV, pp. 701-706, Porto Alegre, Brazil, August 13-18, 1995.
38. Wallin, K., and Rintamaa, R., "Master Curve Based Correlation Between Static Initiation Toughness K_{IC} and Crack Arrest Toughness K_{IA}," Proceedings of the 24th MPA-Seminar, Stuttgart, October 8 and 9, 1998.
39. H. Rathbun, H., Odette, R.G., He, M., Lucas, GH., and Sheckherd, J., "Statistical Versus Constraint Size Effects on the Transition Regime Toughness: A Preliminary Study of a Pressure Vessel Steel,"
40. Gao, X., Dodds, R., Tregoning, R., Joyce, J., and Link, R., "Cleavage Fracture in Surface Cracked Plates: Experiments and Numerical Predictions," *Proc. of the 1999 ASME Pressure Vessel and Piping Conference*, ASME, July 1999.

41. Randall, P.N., "Basis for Revision 2 of the U.S. Nuclear Regulatory Commission's Regulatory Guide 1.99," *Radiation Embrittlement in Nuclear Pressure Vessel Steels: An International Review (Second Volume), ASTM STP-909,* L.E. Steele, Ed., 1987.

42. Kirk, M., Lott, R., Server, W., and Rosinski, S., "Initial Reference Temperatures and Irradiation Trend Curves for use with RT_{To}, A Preliminary Assessment," *Proc. of the 1999 ASME Pressure Vessel and Piping Conference,* ASME, July 1999.

43. Eason, E.D., Wright, J.E., and Odette, G.R., "Improved Embrittlement Correlations for Reactor Pressure Vessel Steels," NUREG/CR-6551, 1998.

44. Lott, R.G., Presentation to the PVRC Working Group on Master Curve, May 1999, Columbus, Ohio.

45. SECY-82-465, Enclosure A, "NRC Staff Evaluation of Pressurized Thermal Shock," November 1982.

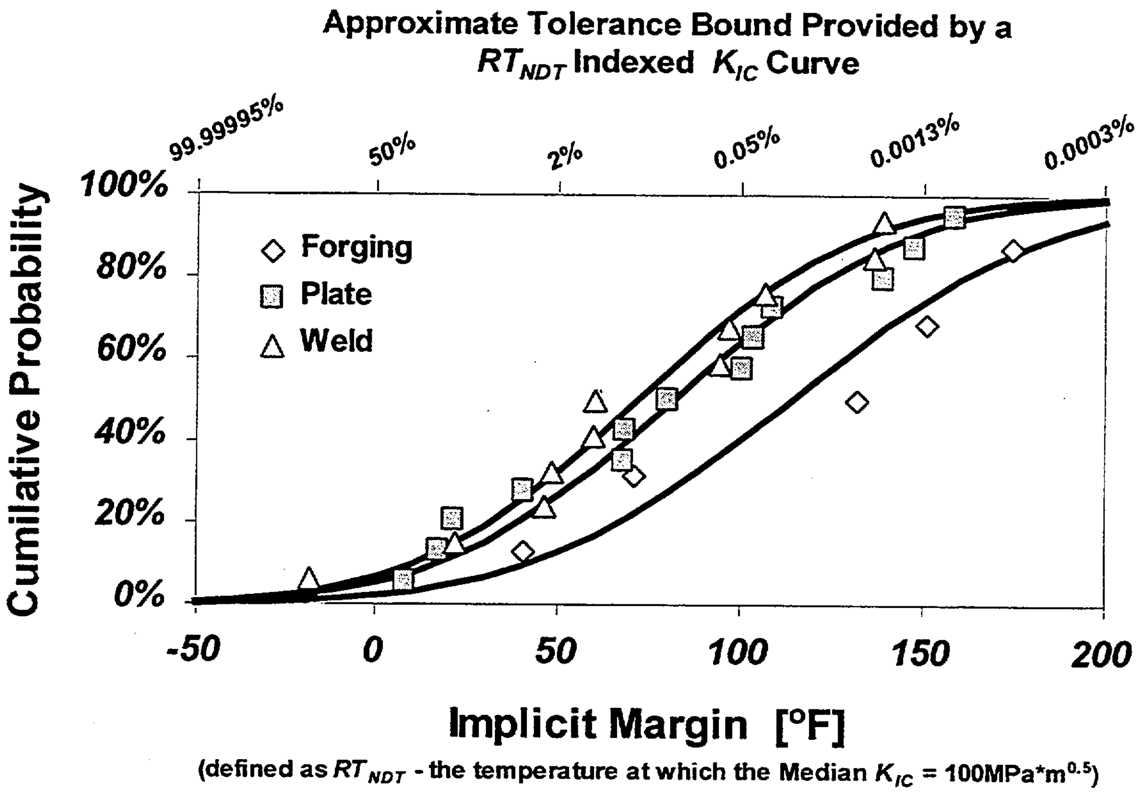

Figure 1. The range of implicit margins (defined as $RT_{NDT} - T_o$) on fracture toughness inherent to a RT_{NDT} indexed K_{IC} curve (data extracted from [29]).

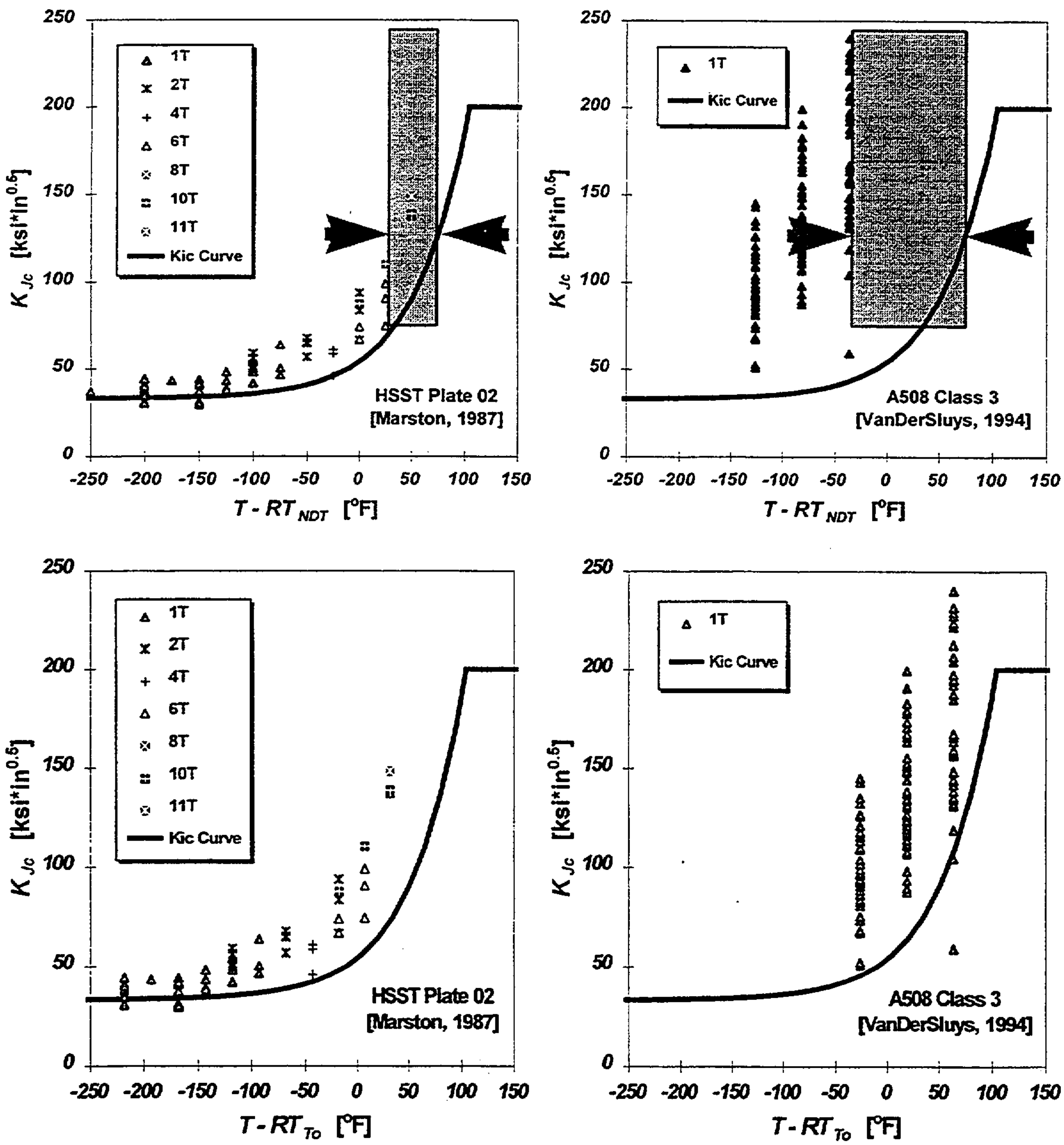

Figure 2. The effect of choice of KIC curve index temperature on implicit margins for A533B Cl. 1 Plate HSST-02 (left) and for an A508 Cl. 3 forging (right). Use of RT_{NDT} (top) produces different implicit margins for different materials, whereas use of RT_{To} (bottom) results in consistent implicit margins.

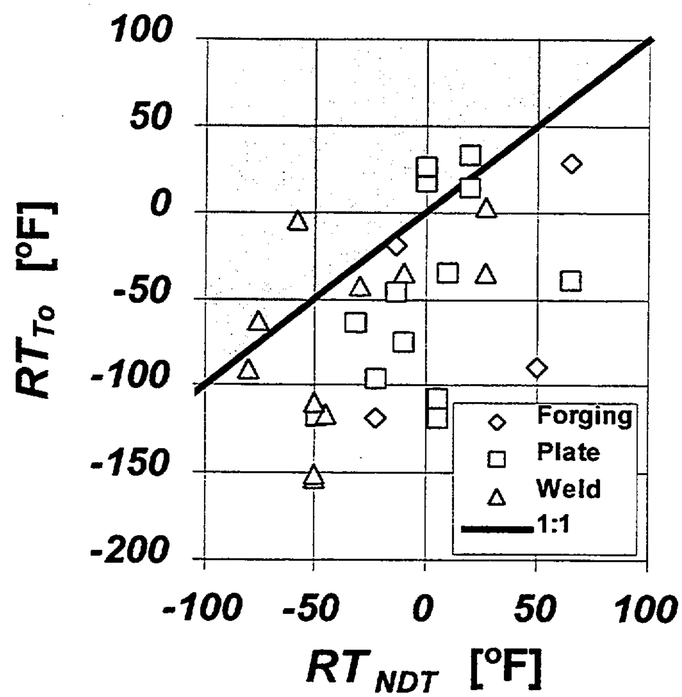

Figure 3. A comparison of RT_{NDT} and RT_{To} values for 26 different unirradiated RPV plates, forgings, and welds (data extracted from [29]). Materials in the shaded region are predicted more conservatively by RT_{To} than by RT_{NDT}.

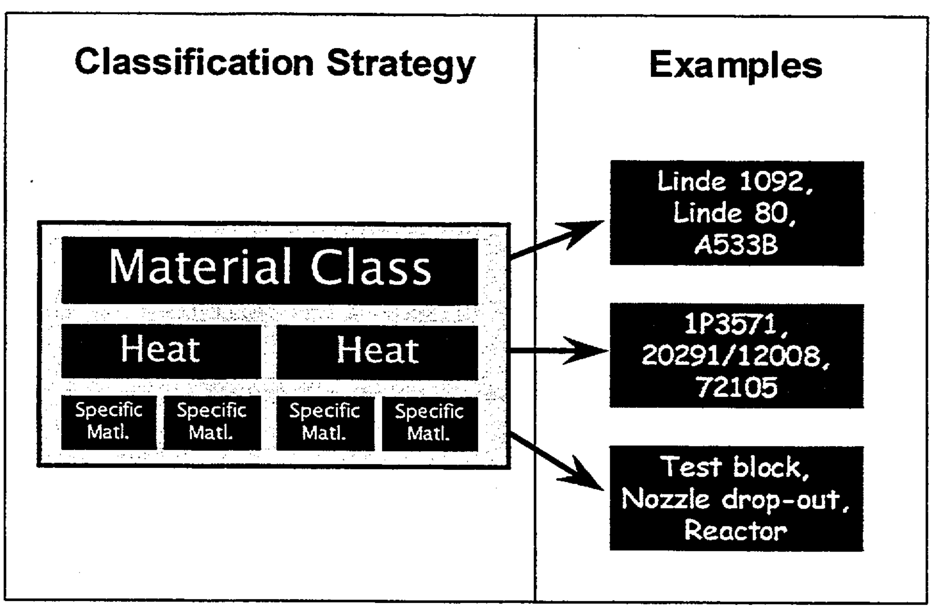

Figure 4. Classification strategy for material uncertainty proposed by Lott [44].

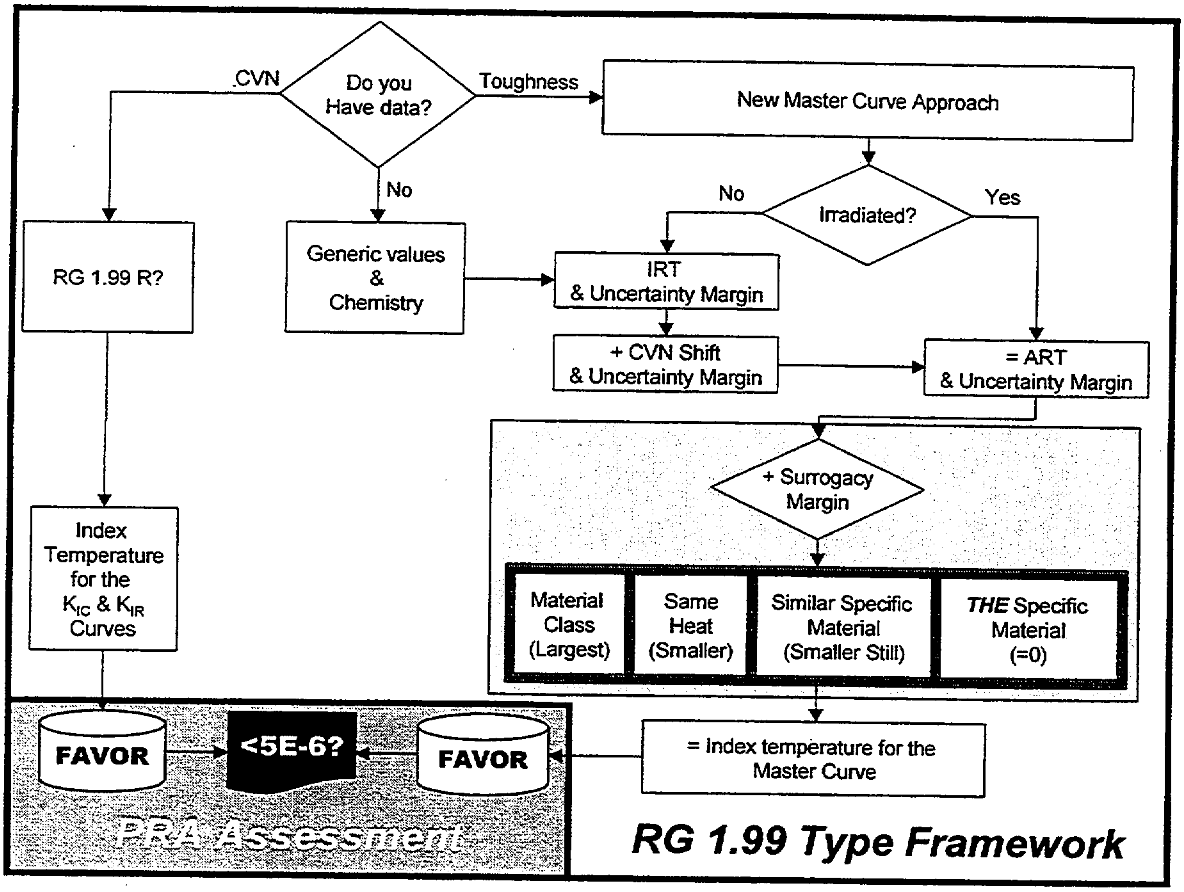

Figure 5. A potential framework for an integrated assessment of the effect of the Master Curve on nuclear RPV integrity assessment. Assignment of margins for surrogacy follow on a classification strategy proposed by Lott [44].

APPLICATION OF PHASED ARRAY UT FOR NUCLEAR PLANT NONDESTRUCTIVE EXAMINATION

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ABSTRACT

EPRI is using phased array technology to develop several ultrasonic inspection techniques for the electric utility industry. The objective is to increase the economic value of the inspections by decreasing inspection time and simplifying the scanning hardware, while matching or exceeding the flaw detection and sizing capabilities of conventional ultrasonic techniques. EPRI is working closely with utilities and NDE vendor companies to develop probes and procedures, to demonstrated and qualify them, to conduct first field application, and to see the array techniques into routine commercial operation. Commercial field deployments in 1999 include inspection of the stainless steel core shroud of boiling water reactors for detection and sizing of intergranular stress corrosion cracking near the welds, and inspection of turbine disks for detection of cracking in the blade attachment hooks. Also under development are rapid-scan techniques for inspection of reactor pressure vessel welds, austenitic welds, ferritic piping welds, and steam generator tubing.

INTRODUCTION

Current ultrasonic examination (UT) techniques require moving one or more transducers across a component's surface in a time-consuming two-dimensional scan pattern. The time required to execute this scan pattern contributes to labor costs, personnel exposure, and in some situations critical path impact. Phased array technology, under investigation by the EPRI Advanced Nondestructive Evaluation (NDE) Program, offers a means to reduce the scanning time by simplifying the scan pattern. As economic pressures force utilities into shorter and shorter outages, the qualified inspection techniques that are employed today may soon prove to be too slow for economic viability.

A single array probe can be made to simulate many different conventional probes. Without moving the probe, sound beams of many angles can be generated sequentially, inspecting a large portion of the component's cross-section. To achieve this same coverage using conventional techniques, the probe would have to be moved over the component's surface. In this manner, a slice of a component may be scanned electronically in milliseconds instead of scanned mechanically in a few seconds. Instead of the slow, two-dimensional scan pattern necessary to scan a weld joint using conventional methods, the probe may simply be swept along the length of the weld once or twice to achieve similar results. In many cases, scan times can be reduced by at least an order of magnitude.

Another advantage of this approach is the possibility of improving inspection coverage in situations of limited access. Even when the available probe positions around an inspection area are limited, high coverage can still be achieved because of the many beam directions that are applied from each probe position. Additional coverage improvements are realized because of the relatively small size of the probe package, as a single array probe may replace several conventional probes. The coverage improvements can be of very high value in key inspection areas of difficult access, such as the welds in a boiling water reactor pressure vessel.

The image produced from this type of polar, electronic sweep is called a "sector scan." Most people already know this type of image; the familiar wedge-shaped, black-and-white ultrasound images of fetuses are sector scans. In fact, the fetal ultrasound imaging technique is almost exactly the same as ultrasonic sector-scan imaging for inspection of power plant components. In addition to medical imaging, phased array technology has been a staple of radar and sonar imaging for decades, as a means for rapid coverage of large areas from a single position. The inspection industry has never had research and development resources approaching the magnitude of those available to the medical and defense industries, but the recent availability of powerful computers and specialized integrated circuits at relatively low cost has enabled the transfer of phased array technology to inspection applications.

There are potential obstacles to implementing these techniques in the field. First of all, the number of NDE vendor companies prepared to offer commercial services including phased array UT in the USA is very limited. Second, the equipment is relatively expensive at this time. It is difficult for an NDE vendor company to invest in phased array equipment and probes without first being assured that there will be a viable market for their use, and that its utility customers will accept the new technology. EPRI's intent is to help make it easier for NDE vendors to equip themselves to offer commercial phased array UT services, by developing and demonstrating qualified, economically valid applications.

The Advanced NDE Technology Program is developing phased array ultrasonic examination techniques for various power plant components, including boiling water reactor core shroud welds, stainless steel piping welds, reactor pressure vessel welds, turbine disc rims, and high-energy piping welds. All of these array applications will benefit from increases in examination speed. Some applications will include particular additional benefits that are made possible by the array technology. For example, using an array to examine the blade attachment hooks of a turbine rotor disc will also provide improved characterization of the configuration of the disc rim, which frequently is not known in advance. Array examination of reactor pressure vessel welds eases discrimination between subsurface flaws and those that are actually surface-connected. This discrimination is important because these two flaw categories have different consequences in terms of the flaw evaluation and the reinspection schedule.

The primary areas of application development in 1999 include boiling water reactor (BWR) core shroud welds; reactor pressure vessel (RPV) welds; turbine disk blade attachment hooks; pipe weld inspection; and austenitic weld inspection. These developments, plus an investigation of phased array inspection for steam generator tubing, will continue in 2000. EPRI is in close contact with several organizations worldwide in order to share information and increase its capabilities in phased array UT.

BWR CORE SHROUD WELDS

The core shroud of a BWR is a stainless steel cylinder surrounding the reactor core. The shroud is about six meters high and about four meters in diameter. It contains several circumferential and vertical assembly welds which are susceptible to intergranular stress corrosion cracking (IGSCC) and therefore must be inspected periodically. A typical shroud inspection would include about 60 meters of weld. The inspections are usually done using ultrasound. Conventional, high-resolution ultrasonic scans of these welds take about 20 minutes per meter of weld. The scan is performed by moving the transducer in a two-dimensional raster pattern over the surface of the shroud adjacent to the weld (Figure 1).

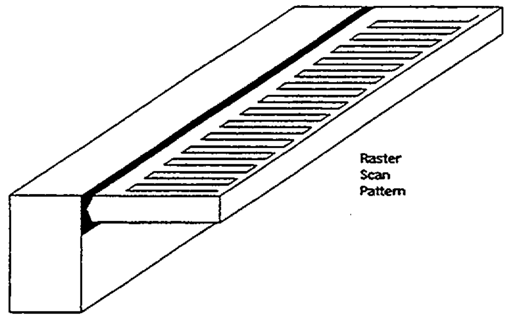


Figure 1. Scan pattern used in conventional inspections of core shroud welds.

By using an array probe, the inspection cross-sectional area of interest can be covered from just three probe positions. The resulting scan pattern (Figure 2) can be executed with a net production speed of about 10 seconds per foot of weld. Since these inspections are performed inside the reactor vessel during refueling outages, the reduction in scan time can be useful in economically scheduling in-vessel activities.

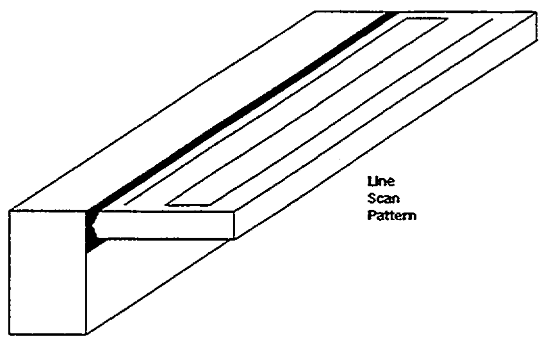


Figure 2. Scan pattern used in array inspections of core shroud welds.

The probe is a 2MHz 2x32-element dual array probe with 22mm aperture. At each data acquisition position, a sector scan of 51 beam angles is generated. The individual beams are from 30° to 80° longitudinal waves, in 1° increments. This small angular resolution was used to achieve optimum depth sizing precision. The depth sizing RMS error was about 1.2mm for cracks on the near side of the weld (sound beams do not have to penetrate weld metal). Even with this fine angular resolution, the data file size was smaller than that experienced using conventional UT techniques, because of the simplified scan pattern.

An example of the ultrasonic response from an unflawed area of a core shroud weld is shown in Figure 3. Figure 3a shows the configuration of the mockup and the placement of the array probe, while Figure 3b shows the sector scan response. The position of the weld is shown in the sector scan with straight white lines. The diagonal fillet of the weld produces a very strong response, while the vertical weld fusion line produces a light, hazy response. The slight tilt and distortion of the sector scan image is characteristic of the imaging software.

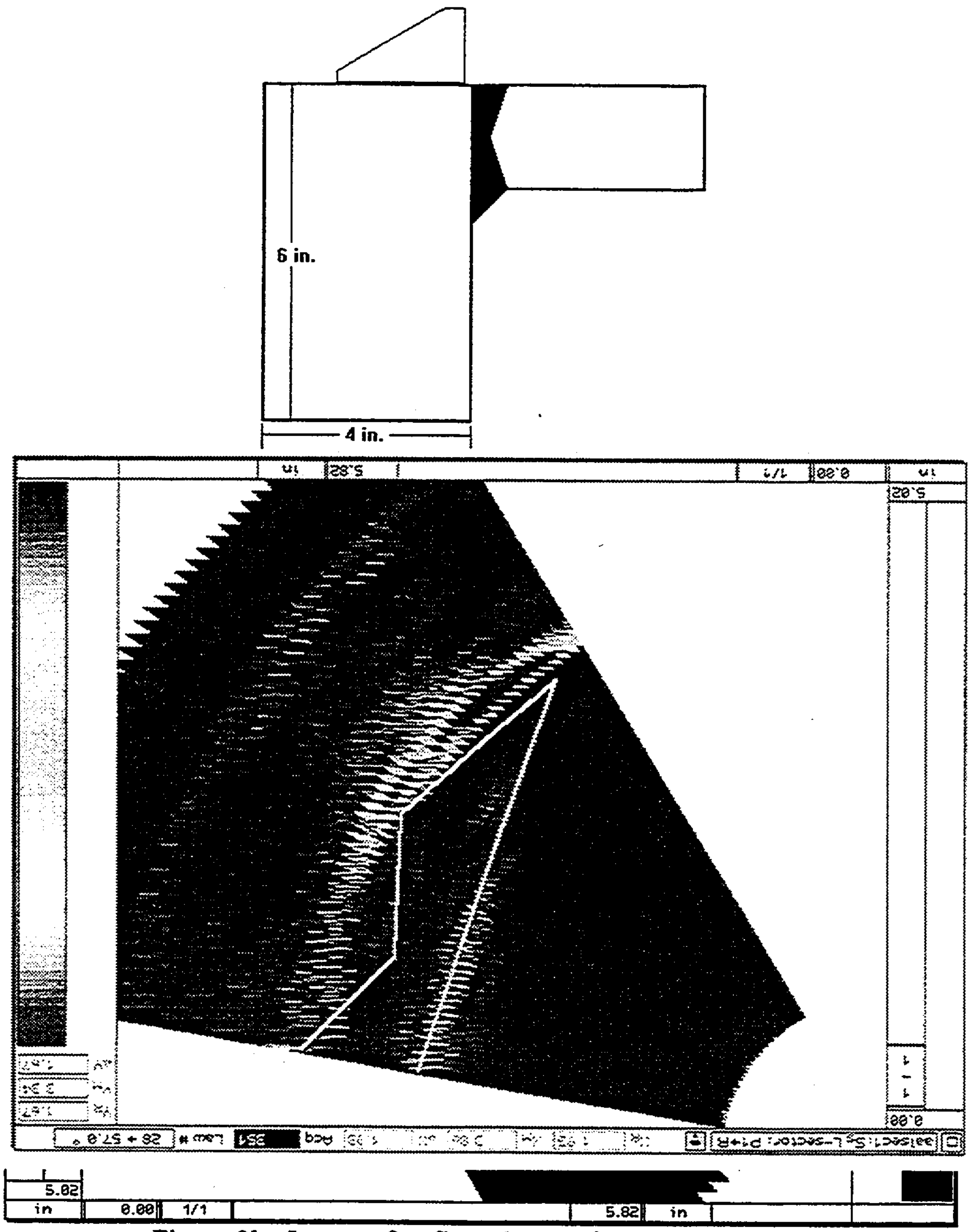


Figure 3b. Image of unflawed area of core shroud weld.

An example of the sector-scan response from a crack connected to the opposite surface is shown in Figure 4. The data analyst is presented with a scaled cross-sectional view of the cross-section of the shroud weld region. His analysis task is reduced to noting the presence of the crack reflection, and measuring the crack depth by applying measurement cursors to the image. The simple and intuitive nature of the data presentation also makes it easier for the analyst to explain his results to the utility customer, and if necessary to the regulator or insurer.

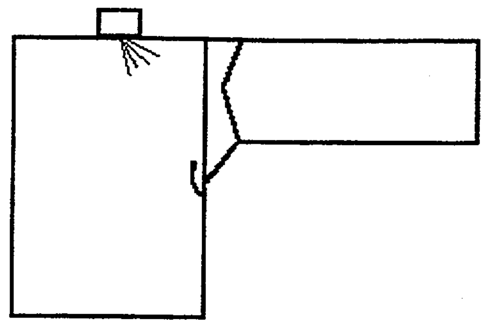
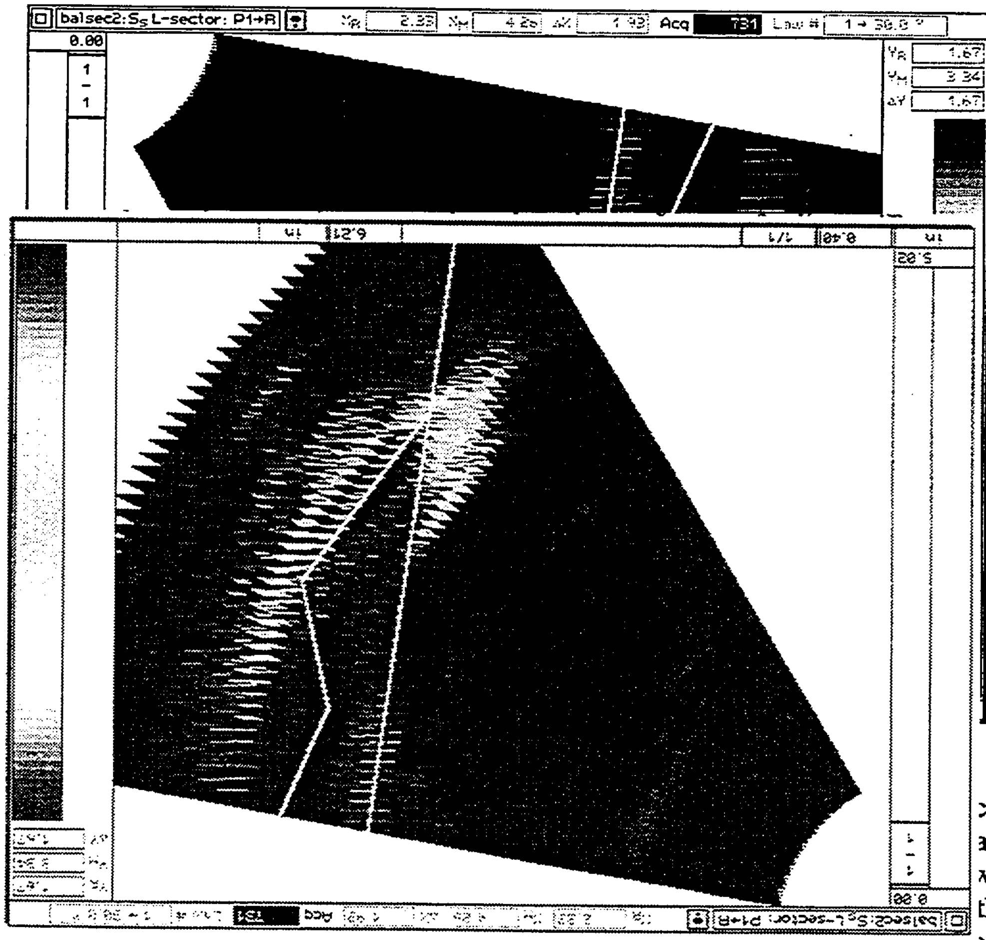


Figure 4a. Core shroud crack connected to opposite surface.



The sectc ... ch faster. Even usii ... akes longer than the a ... veen the time the ı ... the extent of the crack ... give the final results as quickly as possible so that the structural analysis of the shroud can be performed, and run/repair decisions can be made.

This array inspection technique for core shroud welds has been demonstrated using realistic mockups containing cracks that simulate IGSCC (Selby and Romano, 1997). The accuracy of the array technique in measuring crack depth compares favorably with the accuracy of conventional techniques. The first field application of the array technique occurred in May 1998 at a domestic BWR plant. (While this was not the first application of phased-array inspection for the core shroud, it was the first time that the sector scan approach had been used. One core shroud inspection was performed in 1995 using array probes, but the probes were used to produce only a few beam angles. Sector scan imaging was not used, and a conventional, two-dimensional raster scan pattern was performed; therefore, some of the key advantages of phased array inspection were not realized.)

RPV WELDS

After the core shroud, the primary pressure boundary application that is perhaps closest to field deployment is the inspection of the BWR pressure vessel underclad region from the outside surface. Like the core shroud application, this technique uses electronic scanning to cover as much of the inspection area as possible from each probe position, thereby permitting the use of an abbreviated, very fast scan pattern.

The most important region of the pressure vessel to inspect is the inner inch or so of the wall thickness, in and near the welds. This is where cracking is most likely to occur and is also where the stress intensities are high. The inspection area is shown in Figure 5. The industry standard qualification test for this inspection requires the detection of cracks as small as 3mm deep connected to the clad interface. Detection of these small cracks using conventional techniques requires a slow, large-area scan and careful analysis of the data; the signal-to-noise ratio of these small cracks is sometimes low, in part because the sound beam has spread so much by the time it reaches the crack.

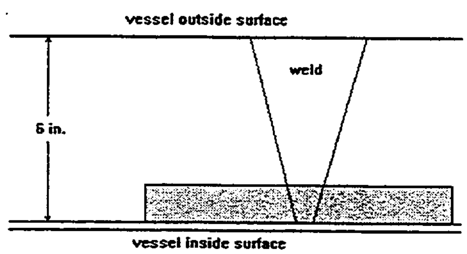


Figure 5. Inspection area for BWR vessel inner surface.

EPRI is using a 4MHz, 50mm dual (separate transmit/receive arrays) array probe to generate longitudinal waves that are focused at the inside surface. Because the beam is focused, a greater proportion of the sound energy is returned from the small crack and the signal-to-noise ratio is improved. Without moving the probe, the position of the focal spot is swept along the inside surface in 5mm increments over a range of 300mm (Figure 6). The entire inspection area can be covered from just two probe positions. The probe is scanned parallel to the weld, moved away from the weld by 100mm, and scanned parallel to the weld again (Figure 7). The scan is repeated from the other side of the weld, or else two array probes are used and both scans are done simultaneously. The time saving is even greater than that shown for the core shroud weld, because the conventional scan pattern for the pressure vessel is larger and slower.

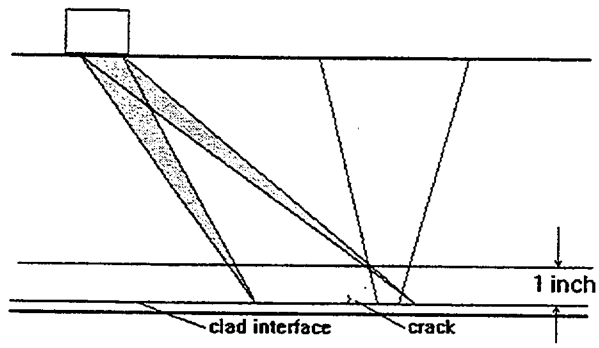


Figure 6. Using focusing to inspect the vessel inner surface.

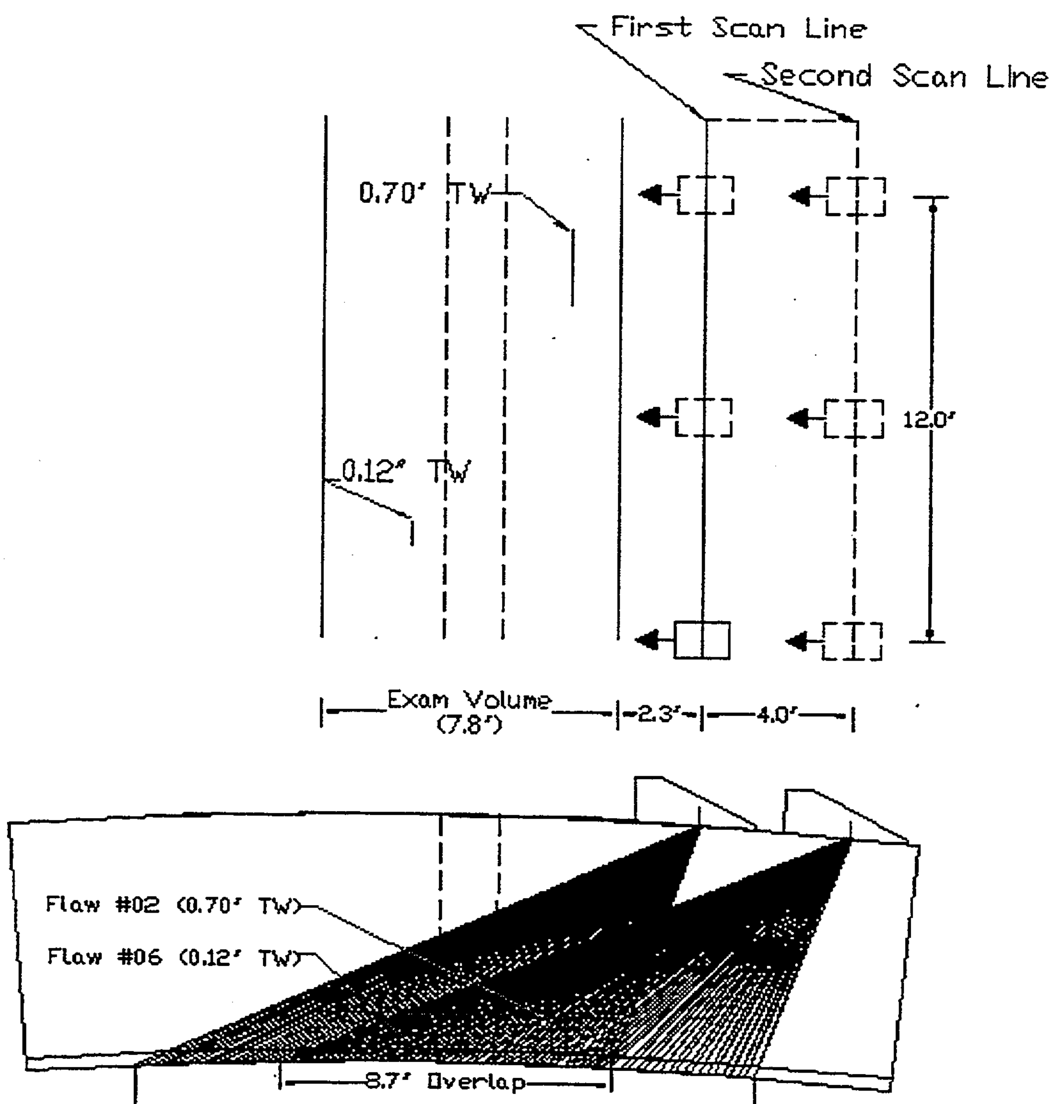


Figure 7. Scan pattern (one direction) for rapid examination of BWR RPV welds underclad region.

This technique was used to examine an RPV plate mockup containing eight underclad cracks. All eight cracks were detected, and all eight cracks were shown in each of the four scan lines (two scan lines for each beam direction). The response of a 3mm deep underclad crack is shown in Figure 8. The clad response is visible as a horizontal band in the sector-scan image; the crack response is just above it, in the center of the image. Slightly to the left of the crack response is a small group of responses from unintentional reflectors associated with fabrication of the underclad crack. The A-scan shows the maximum response from the crack, illustrating the excellent signal-to-noise ratio.

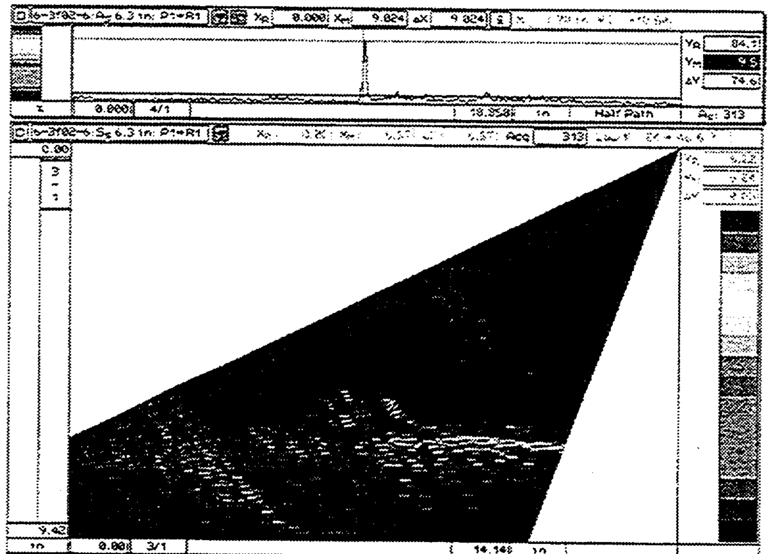


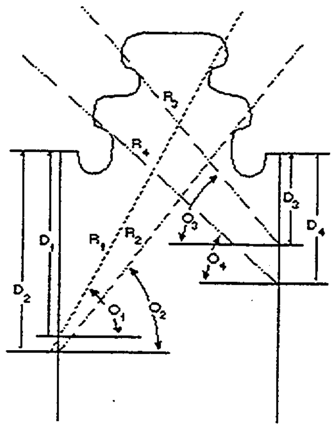
Figure 8. Sector-scan image of a 3mm deep underclad crack. The A-scan is the response from the crack, in this case using a beam angle of 46.6 degrees

Inspection vendor companies are already interested in applying this technique in the field, and a candidate demonstration plant will be sought after a qualification is complete.

TURBINE DISK RIM BLADE ATTACHMENT HOOKS

EPRI has developed a method for using array probes to speed the inspection of turbine disk rims. These GE-type disk rims are geometrically complex and are difficult to inspect using conventional probes, because the configuration of the rim is sometimes not known. Array imaging can be used to compensate for the uncertainty of the part geometry. EPRI worked with a US inspection vendor to refine the array inspection techniques, and the first field application is scheduled for October 1999.

The conventional UT technique inspects each attachment hook with a dedicated probe of a carefully selected beam angle. The probe must be held at the correct radial position as a circumferential scan is performed. If the radial position is allowed to drift, the scan must be repeated. This process is repeated until all hooks have been examined. See Figure 9 for a schematic of the conventional technique.



Location	Met. Dist.	Dist. Tang
θ_1	R_1	D_1
θ_2	R_2	D_2
θ_3	R_3	D_3
θ_4	R_4	D_4

Figure 9. Conventional UT for turbine disk rims. A separate probe is used for each hook.

The phased array technique employs a sector scan to examine all the hooks simultaneously, as shown in Figure 10. The technique is not very sensitive to radial position of the probe, and only one circumferential scan is needed on each face of the disk. Figure 11 shows a sector scan of a three-hook disk rim with a crack in the first hook. The rim geometry has been overlaid on the image for clarity. The presence, location, and nature of the defect indication are unmistakable. This image was obtained using a 10MHz, 32-element array of 10mm aperture, generating a longitudinal-wave sector scan of 0.5° resolution.

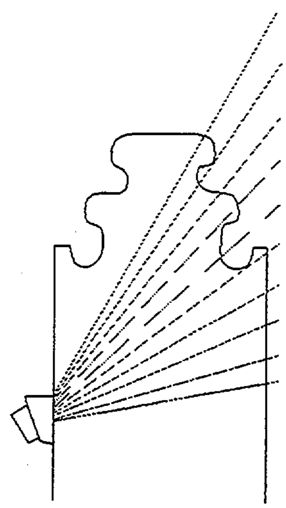
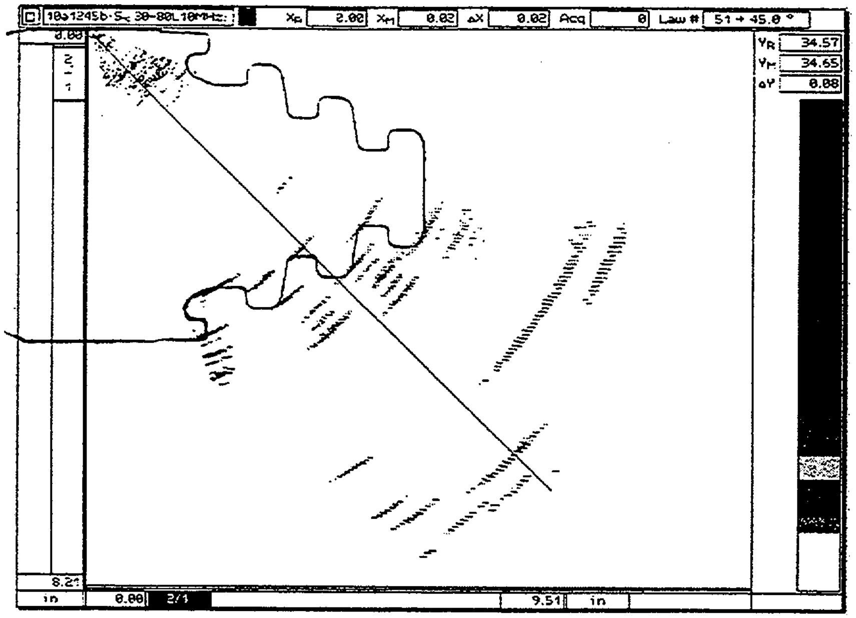


Figure 10. Sector scan of all hooks on one side simultaneously.



Specimen A12, crack area 2
~35-40° angle to flaw in first hook
10mhz, 32 element array
20° - 70° longitudinal waves, 0.5° increment

Figure 11. Sector scan of a three-hook rim with a crack in the first hook.

The UT data analysis software can import CAD drawings of the rim geometry to aid interpretation. In cases where the geometry is not known, EPRI has shown that the sector-scan data can be used to reconstruct it.

In 2000 EPRI will develop inspection techniques for the even more complex Westinghouse-type, axial-entry disk rims.

HIGH-ENERGY PIPING WELDS

EPRI is developing a rapid-scan inspection technique using arrays for high-energy steam piping. This piping can fail by creep fatigue cracking along the longitudinal seam welds. Fossil power plants may contain hundreds of feet of this piping, so inspection of the welds is time-consuming. In present-day inspections, a rapid scan is performed to detect degradation, but if any degradation is detected, a very slow re-scan must be performed to characterize it. The EPRI technique under development shows promise for collecting characterization data at the same time as the detection data is acquired, greatly reducing the time and cost of the inspection.

AUSTENITIC WELDS

EPRI is investigating inspection of austenitic pipe welds, particularly for IGSCC, using electromagnetic-acoustic transducers (EMATs), and is also investigating detection and sizing of fatigue cracking inside austenitic weld metal using piezoelectric arrays.

EMAT inspection of piping welds

EPRI is also developing inspection techniques for stainless steel pipe welds. In some cases, these techniques employ EMATs to generate a special mode of ultrasonic beams. These horizontally polarized shear (SH) beams can penetrate stainless steel welds much better than conventional modes of ultrasound, which gives EMAT inspection an advantage when a weld can be inspected from only one side. The capabilities of EMATs to satisfy the requirements of ASME Section XI Appendix VIII, and significant advantages for flaw detection and length sizing through the weld, have been demonstrated. Efforts to extend that capability to detection of IGSCC have not been as promising. EPRI is investigating the possibility of using higher-frequency or more efficient EMAT probes to improve the detectability of IGSCC.

Piezoelectric inspection of fatigue cracks in austenitic weld metal

EPRI is investigating the detection and sizing of fatigue cracks in austenitic weld metal. Three cracked specimens have been made from welded 316L stainless steel plates 50mm thick. This work is being conducted in cooperation with Electricité de France.

One large plate was sectioned into four specimens (Figure 12). Each specimen received an EDM notch at the edge of the weld root to serve as a crack starter, and then was fatigued to grow a crack (Figure 13). One specimen was fatigued until the crack depth visible on the side of the specimen was about 15mm; the specimen was then heat-tinted, and then fatigued until failure. Figure 14 shows the shape of the crack and the EDM starter notch. The crack depth is fairly uniform, but not perfectly constant, across the 150mm width of the specimen. The remaining three specimens were fatigued until the visible crack depth at the edges was 5mm, 10mm, and 15mm respectively.

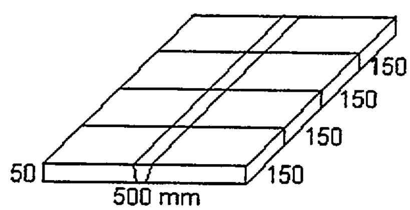


Figure 12. Sectioning of 316L plate to make four fatigue specimens.

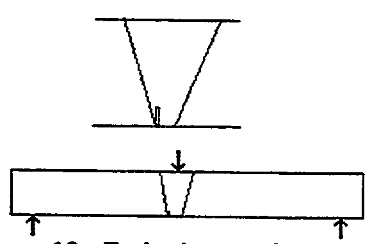


Figure 13. Fatiguing configuration.

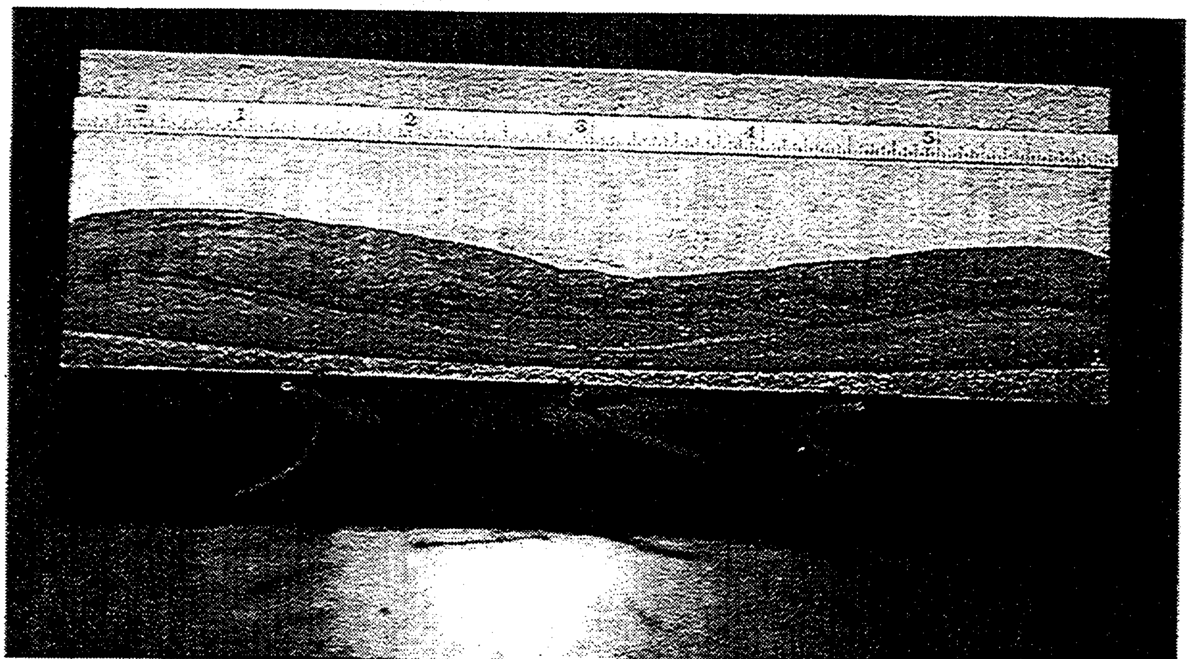


Figure 14. Exposed face of crack that was grown to 15mm depth, heat-tinted, and then fatigued to failure. The slightly curved, 6mm EDM starter notch is visible at the base of the crack.

Each cracked specimen was scanned using a 2MHz, 2x32-element dual array probe with 22mm aperture. At 5mm increments along the weld length, a 51-angle sector scan was obtained. Each sector scan included longitudinal-wave beam angles from 30° to 80° in 1° increments. (This is the same probe and sector scan scheme that was described for BWR core shroud welds.) The weld was scanned from both directions (Figure 15).

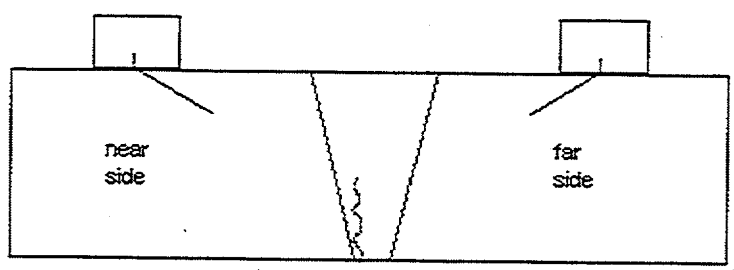


Figure 15. Scan directions for fatigue plates.

Figure 16 shows a graph of the crack depth measurements for one of the specimens. The measured crack profile is in good agreement with the two crack depths observed on the sides of the specimen, and with the crack profile shape that was expected based on the shape of the trial crack and the shape of the EDM starter notch. Also, the near-side measurements usually agree well with the far-side measurements.

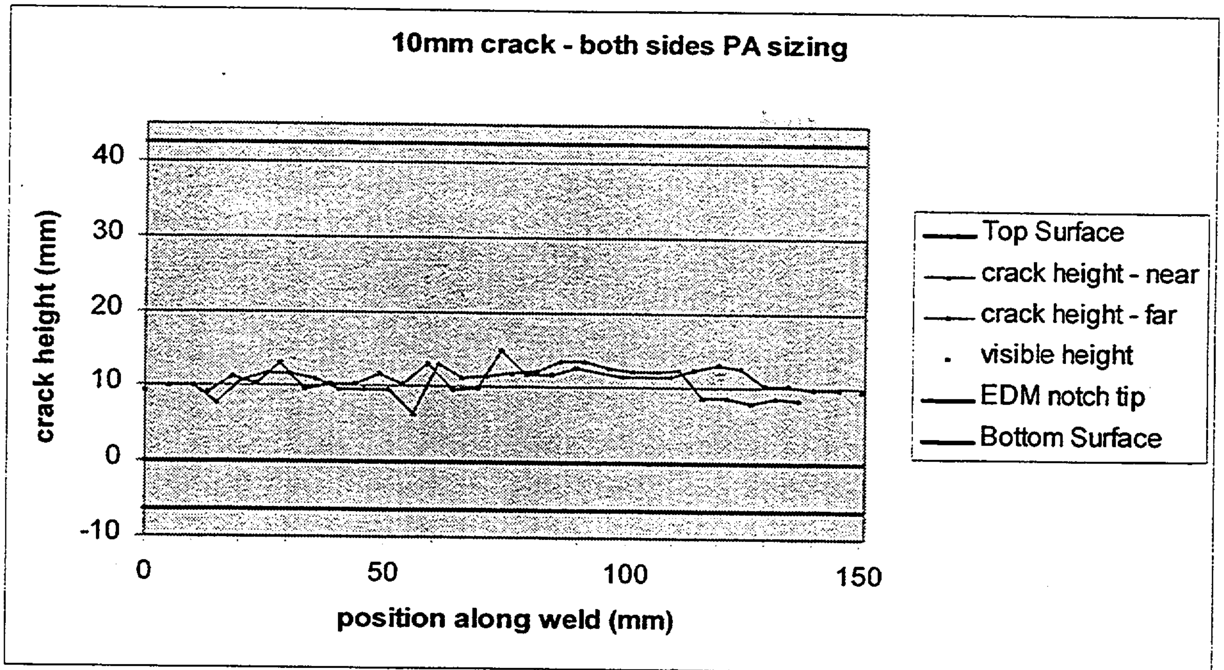


Figure 16. Graph of UT crack depth measurements along one of the fatigue cracks.

The next steps in this program will include the removal of the EDM starter notches, rescanning the specimens, and comparison with the responses of EDM notches in the weld of a similar specimen.

INSPECTION OF TUBING

High-speed phased array inspection of steam generator tubing has been developed and demonstrated outside the United States. EPRI hopes to demonstrate similar performance using equipment readily available to domestic inspection vendor companies, and support field applications.

PRESSURE VESSEL NOZZLE INNER RADIUS AND NOZZLE-TO-SHELL WELDS

These geometrically complex inspections require the use of many specially-designed probes to achieve adequate coverage. Array probes could be used to drastically reduce the number of probes necessary, the inspection time, and/or the mechanical complexity of the scanning equipment. In 2000 EPRI will lay the groundwork for formal demonstration and field deployment of array inspection techniques for pressure vessel nozzles.

CONCLUSION

The use of phased array ultrasonic techniques can improve on the performance of conventional inspection techniques in many ways. Improvements can include reduction in inspection time, reduction in data analysis time, improvements in ease and reliability of data analysis, reduction in the mechanical complexity of scanning equipment, and improvements in the reliability of detecting small defects.

EPRI is simultaneously developing phased array ultrasonic inspection techniques for several power plant components. In some cases, phased array applications have been developed outside the United States, but few are used routinely and very few have been applied in the United States even once. EPRI believes that, with the capabilities of industrial phased array equipment improving and the cost decreasing with each product generation, very soon vendors will be able to offer economically attractive phased array solutions to inspection problems. EPRI will have several applications already developed, demonstrated, and deployed in field trials in order to smooth and speed the process of making these solutions available to its utility customers.

REFERENCES

1. SELBY G P and ROMANO S R, *Procedure for Phased Array Ultrasonic Examination of Boiling Water Reactor Core Shroud Welds*, EPRI GC-108070, Electric Power Research Institute, December 1997.

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S. Nesmith, NRC Project Manager; Transactions prepared by Brookhaven National Laboratory

11. ABSTRACT *(200 words or less)*

This contains transcripts of papers presented at the Twenty-Seventh Water Reactor Safety Information Meeting held at the Bethesda Marriott Hotel, Bethesda, Maryland, October 25-27, 1999. The papers describe the programs and results of nuclear safety research sponsored by the Office of Nuclear Regulatory Research, U.S. NRC. Papers concerning nuclear safety issues from U.S. government laboratories, the electric utilities, the nuclear industry, and from foreign governments and industry are also included. They have been compiled in one report to provide a basis for meaningful discussion and information exchange during the course of the meeting, and are given in the order of their presentation in each session.

The papers are printed in the order of their presentation in each session and describe progress and results of programs in nuclear safety research conducted in this country and abroad. Foreign participation in the meeting included papers presented by researchers from Belgium, Finland, France, Japan, Russia, and Switzerland. The titles of the papers and the names of the authors have been updated and may differ from those that appeared in the final program of the meeting.

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